NUCLEAR SCIENCE ABSTRACTS

Volume 11 July 31, 1957 No. 14 Suppl.

GENERAL

ATOMIC POWER

Refer to abstract 7817.

BIOLOGY AND MEDICINE

7398 UCLA-329
California. Univ., Los Angeles. Atomic Energy Project.
QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1955. Apr. 10, 1955. Decl. Mar. 6, 1957. 111p. Contract AT-04-1-GEN-12. $16.80(ph OTS); $5.70(mf OTS).

Progress is reported in studies on the metabolism of linoleic acid in rats; the isolation of high-molecular-weight nucleic acids; the physical properties of biological macromolecules; the synthesis and analysis of unsaturated fatty acids; the central and peripheral effects of injected chlorpromazine in dogs; the effects of total-body irradiation on the excretion of histamine by rats and on the concentration of blood plasma Cu in burros; the protective effects of reticulo-endothelia system stimulants against acute whole-body x irradiation in mice, rats, and rabbits, using undulant fever vaccine, antipyretics, and sodium salicylate as test materials; the development of labeled material for use in tests of the phagocytic function of the reticulo-endothelia system; external γ-ray scintillation counting applied in the evaluation of bilateral kidney function following the intravenous injection of minute quantities of I^{131}-labeled Urokon; a method for the determination of Sr^{90}–Y^{90} and Ca^{45} in the same sample; the effect of sulfate, acid phosphate, phosphate ions, Ca, and Sr concentration on skeletal deposition of Sr^{90}; the production, isolation, and assay of an antibiotic-like agent produced by a lactobacillus isolated from the rat's small intestine; the late effects of irradiation on blood pressure in rats; radioinduced changes in the lipids and lipoproteins of embryo plasma; the spectrographic determination of Sr and Ca in plant ash samples and soil leachate ash samples; the preparation of samples for electron microscopy and methods for cleaning the specimen screens; the availability of Fe^{59} incorporated in glass-like materials to various crop plants; the effect of environmental factors on plant uptake of radioactive materials under controlled conditions; and studies associated with the development of instruments for measuring fast neutron flux and the development of chemical radiation detectors. (C.H.)

RADIATION HAZARDS AND PROTECTION

7399 ANL-5289
Argonne National Lab., Lemont, Ill.
ENVIRONMENTAL RADIOACTIVITY AT ARGONNE NATIONAL LABORATORY. Report for the Year 1953. J. Sedlet and A. F. Stehney. Decl. Feb. 14, 1957. 35p. Contract W-31-109-eng-38. $6.30(ph OTS); $3.00(mf OTS).

Data are summarized on the radioactive content of samples of rain, surface water, soil, plants, and bottom silt from the beds of surface waters which were collected and analyzed during 1953. (For preceding period see ANL-5669.) (C.H.)

7400 KAPL-863
Knolls Atomic Power Lab., Schenectady, N. Y.
RADIOLOGICAL DEVELOPMENT ACTIVITIES IN THE HEALTH AND SAFETY UNIT. Semiannual Progress Report for July–December 1952. Jan. 1953. Decl. Feb. 7, 1957. 80p. Contract W-31-109-Eng-52. $9.30(ph OTS); $3.60(mf OTS).

Progress is reported on separations pilot plant stack effluent studies, determination of the counting efficiency of an inert gas monitor, evaluation of the neutron spectra external to KAPL reactors, development of a BF_3 tube fitted with concentric polystyrene cylinders of varying thicknesses for intermediate neutron monitoring, and development of a procedure for the analysis of radiostrontium in vegetation. The particle-collection efficiency of an electrostatic precipitator and glass fiber air-filtering unit and performance characteristics of standard laboratory fume hoods under operating conditions were determined. A new sampling unit was designed for the collection of air-borne particulate matter, and data on its performance are included. It was concluded that Vinylite plastic bags are the most satisfactory method for protecting survey instruments from radioactive contamination. The maximum permissible air concentration for Th^{232} was calculated to be in the range of 1 to 7×10^{-12} μc/cc. It was concluded that incidence of biolobed nuclei in lymphocytes is not a practical indication of radiation exposure. (C.H.)

CHEMISTRY

7401 A-1247
Columbia Univ., New York. Div. of War Research.
ATTEMPTS TO CONSOLIDATE AMORPHOUS BORON. Helen H. Starke. Mar. 10, 1944. Decl. Feb. 12, 1957. 27p. Contract W-7405-Eng-50. (100B-R-82). $6.30(ph OTS); $3.00(mf OTS).

Amorphous boron materials were fused into microcrystalline products high in boron by induction heating in a boron carbide container in a He atmosphere. Resistance heating did not produce a sufficiently high temperature for fusion. (C.W.H.)

7402 A-1293
Wisconsin. Univ., Madison.
FRACTIONATION AND SIZE DISTRIBUTION ANALYSIS OF NICKEL CARBONATE PREPARATIONS. Final Report. E. M. Bevilacqua, L. J. Gosting, and J. W. Williams. May 1, 1944. Decl. Feb. 12, 1957. 29p. [For Columbia Univ. Div. of War Research.] Contract W-7405-eng-50, Subcontract No. 13. (100XR-5610). $4.80(ph OTS); $2.70(mf OTS).

7403 A-3111
American Cyanamid Co. Stamford Research Labs., Stamford, Conn.
PRODUCTION OF 891. Progress Report for August 16 – October 1. J. K. Dixon. Oct. 1, 1944. Decl. Feb. 12, 1957. 44p. Contract W-7401-eng-91. $7.80(ph OTS).

A process was developed for the production of 600 g/day

from dimethyl ether boron trifluoride complex of crystalline B of better than 98% purity, with a fineness of 200 mesh or better. (J.E.D.)

7404 ANL-4265
Argonne National Lab., Lemont, Ill.
MINUTES OF CONFERENCE ON THE CHEMISTRY OF RUTHENIUM, DECEMBER 13 AND 14, 1948. H. Baxman, E. Turk, and L. Deutsch, comps. Decl. Feb. 14, 1957. 56p. Contract W-31-109-eng-38. $9.30(ph OTS); $3.60 (mf OTS).

The conferees discussed gravimetric, colorimetric, and radiochemical techniques for Ru analysis. Acceptable procedures were reported using each of these techniques. The determination of valence states was in a less satisfactory state. Ruthenium tetroxide and ammonium hexachlororuthenate were suggested as primary standards. The absorption spectrum of Ru in a wide variety of solutions has been investigated and some peaks identified as characteristic of a particular Ru structure in different solutions. The effects of complexing ions, acidity and valence states have all been studied by this technique. Polarographic data of a preliminary nature were presented for various solutions. The difficulty of analyzing typical solvent extraction data on the behavior of Ru was emphasized. The possibility of radiocolloid formation as an explanation was mentioned. It was pointed out that other platinum metals showed similar irreproducibility in extraction into hexone from dissimilar aqueous solutions. The behavior of Ru tetroxide in solvent extraction was discussed as well as the production and volatilization of this form of the element. Its use as a special decontamination procedure was emphasized. (auth)

7405 ANL-4564
Argonne National Lab., Lemont, Ill.
CHEMISTRY DIVISION, SECTION C-11 SUMMARY REPORT FOR OCTOBER, NOVEMBER AND DECEMBER 1950. J. R. Gilbreath and O. C. Simpson, Comps. May 11, 1951. Decl. Feb. 12, 1957. 124p. Contract W-31-109-Eng-38. $19.80 (ph OTS); $6.30(mf OTS).

Changes in resistance of the "c" axis of high-quality natural graphite crystals from Ticonderoga, N. Y., on irradiation in the X-10 converter for 36.5 Mwd are tabulated. Gold-contact and mercury-contact apparatus for measuring potential differences along the "a" axis are illustrated, and "a" axis resistivities of the unirradiated, previously mentioned crystals are listed. A table of damage to the electrical conductivity of spectroscopic graphite rods irradiated in the CP-3' pile and a graph showing effect of pile irradiation on the magnetic susceptibility of graphite are given. An attempt is being made to detect free radicals formed by γ (Co^{60}) irradiation of ice at liquid-N_2 temperature; at this temperature the radicals remaining frozen into position. Experiments on detection of these "frozen" radicals by a paramagnetic method and by determination of Fe^{+3} in Fe^{++} solution oxidized by the free radicals are described briefly, and some tentative conclusions concerning the negative or inconclusive results obtained are drawn. Data on the far-ultraviolet absorption bands observed in KI crystals containing color centers are summarized. Changes in the absorption spectra of KI due to temperature, photochemical coloration, and additive coloration are illustrated. The origin of the absorption bands is discussed, and some observations on the purity of the KI used are given. Samples of synthetic NaCl and natural rock salt have been compared with respect to behavior of their F-centers produced by exposure to soft x rays; the results are presented in some detail. Preliminary results on the thermal self-diffusion of Ag in the temperature range 785 to 895°C are presented in tabular and graphical form. The Ag was irradiated for 36 hr in the CP-3' thimble, and its self-diffusion was followed by an autoradiographic technique. As part of a detailed study of the mechanism of oxidation of formic acid by γ rays in dilute aqueous solution, a method for measuring the proportion of radicals undergoing recombination and those available for reaction with formic acid is described. The effects of physical treatment of the irradiation cells and purity of the water on reproducibility of results in studies on the decomposition of aqueous H_2O_2 by γ rays are tabulated. The effect of pH on the γ-ray-induced exchange of D_2 gas with liquid H_2O and the yield of ferric ion in the oxidation of ferrous sulfate in H_2SO_4 by β rays from tritium disintegration are plotted. A careful search for delayed coincidences in Ag^{111} decay, using liquid-N_2-cooled stilbene scintillation counters, has confirmed the identification of the 243-kev level in Cd^{111} with the 0.09-μsec transition observed in In^{111} decay and in the 48-min Cd^{111m} transition. Preliminary experiments have been completed on a single-channel differential pulse analyzer to be used in determining $\beta-\gamma$ angular correlation as a function of the β energy. The effect of light collection efficiency on the pulse height and line width from an anthracene crystal, a comparison of pulse height and line width of pulses from an anthracene and a stilbene crystal, and Tm^{170} pulse-height distribution in a NaI(Tl) crystal are presented graphically. A self-delayed coincidence circuit using a NaI(Tl) crystal is described which practically eliminates the background caused by photons originating in the light guide of a scintillation counter of the type used in a double-lens spectrometer and which is expected to improve the low-energy efficiency of the counter. The reproducibility of the isotopic assay of $U^{235}-U^{238}$ mixtures by a photomultiplier technique, using the spectroscopic pair 4244.12–4244.37 A, is indicated in tabular form. A method for making comparator settings photoelectrically which considerably facilitates wavelength measurements of spectrum lines on a photographic plate is described, and the advantages of photoelectric scanning over visual setting by a microscope cross hair are listed. Separation of Na from 100 times its weight of Ca phosphate in the analysis of bone for Na has been accomplished by addition of excess $FeCl_3$ followed by precipitation of the phosphate and excess iron with NH_4OH. A convenient mercury-cathode cell which is simple and easy to operate is illustrated. (auth)

7406 ANL-4888
Argonne National Lab., Lemont, Ill.
CHEMISTRY DIVISION, SECTION C-II SUMMARY REPORT FOR APRIL, MAY, AND JUNE 1952. J. R. Gilbreath and O. C. Simpson, comps. Jan. 19, 1953. Decl. Feb. 12, 1957. 72p. Contract W-31-109-eng-38. $12.30(ph OTS); $4.50(mf OTS).

Progress is reported on the following problems: physical properties of graphite; effects of irradiation on the properties of graphite, on ceramic materials, on ice, and on the decomposition of nitrate and chlorate crystals; investigation of optical properties of single crystals; radiation chemistry of aqueous solutions; application of mass spectrometry to the study of oxygen exchange with water; angular correlation and intensity of inner bremsstrahlung; half life of the metastable state in Np^{237} arising from alpha decay of Am^{241}; gamma radiation from Th^{233}; the high-energy β transition in Cs^{137}; spectrographic analysis of fine structure of T_2, of the spectrum of Am, and of relative abundance in Li^6-Li^7 samples; spin-spin interaction in HD nuclear resonance; and testing of the 60-inch cyclotron. (C.W.H.)

7407 ANL-5466
Argonne National Lab., Lemont, Ill.
CHEMICAL ENGINEERING DIVISION SUMMARY REPORT [FOR] APRIL, MAY, AND JUNE 1955. July 27, 1955.

Decl. Feb. 7, 1957. 92p. Contract W-31-109-eng-38. $0.60(OTS).

Progress is reported on the following investigations: direct-cycle boiling-reactor entrainment studies; dissolution of U, Zr, and Al in BrF_3 solutions; volatilization of U from UF_4 with O_2; corrosion of Ni in fused-salt solutions; kinetics of the U–BrF_3 reactions; development of a vacuum casting furnace for slagging; extraction of Pu and U with liquid Mg; corrosive effects of molten U and Mg on container materials; oxidative slagging process for purification of spent fuel elements; denitration of uranyl nitrates; effect of reduction step on hydrofluorination of UO_2; calcination of fuel-processing waste solutions; spectrophotometric determination of Pu; and volatilization of UF_6 from KF–BrF_3 systems. (C.W.H.)

7408 ANL-5560
Argonne National Lab., Lemont, Ill.
CHEMICAL ENGINEERING DIVISION SUMMARY REPORT [FOR] JANUARY, FEBRUARY, AND MARCH 1956. May 2, 1956. Decl. Feb. 12, 1957. 139p. Contract W-31-109-eng-38. $0.80(OTS).

Measurements of radioactive carryover were made on the Borax III boiling reactor. Mass transfer in a continuous-flow mixing chamber was investigated; additional light scattering data were obtained. The evaluation of Ni in fluorinated and hydrofluorinated fused salts was continued. Corrosion rates of several metals and alloys in BrF_3–UF_6 systems and gaseous F_2 were measured. The radiochemical properties and volatilization of fission product fluorides (TeF_6) were investigated. Studies were continued on the preparation, handling, and thermal stability of PuF_6. Uranium tetrafluoride produced from ore concentrate was fluorinated in a fixed-bed reactor to produce volatile UF_6. Melt-refining of spent EBR fuel pins and the corrosive effects of molten U on alumina and Ta were investigated. Additional studies were conducted on the extraction of Pu from U with liquid Mg, and the subsequent distillation of the Mg to recover the Pu. Several methods of inert gas purification were tested. Procedures are described for the direct assay of Mo and Ru in ternary U alloys by x-ray emission. (For preceding period see ANL-5529.) (C.W.H.)

7409 BMI-1022
Battelle Memorial Inst., Columbus, Ohio.
DISSOLUTION OF ALUMINUM-CANNED THORIUM. John G. Beach, William C. Schickner, and Charles L. Faust. July 26, 1955. Decl. Feb. 13, 1957. 16p. Contract W-7405-Eng-92. $0.25(OTS).

From the standpoint of dissolution characteristics, single-dip-canned Th slugs present a system of four components based on dissolution characteristics: Al, AlSi, and $ThSi_2$ coatings, and the Th core. For reprocessing by the Thorex process, irradiated slugs are dissolved in a mixture of HNO_3 and HF. Dissolution of the Th core, however, is prevented by the $ThSi_2$ layer, which is not attacked by the acid mixture. The acid mixture corrodes type 309 stainless steel equipment and introduces Cr into the reprocessing solutions. Chemical dissolution of the slug components was slow. Electrolytic dissolution with coated Th as the anode resulted in dissolution rates for the various components several times those rates obtained with simple chemical dissolution. The $ThSi_2$ layer was easily removed. Nitric acid with $Hg(NO_3)_2$ but without HF dissolved the Al can but not the AlSi. Thorium silicide and Th were disintegrated by anode electrolyses in HNO_3 alone. In HNO_3–HF, Th dissolved but some of the Th precipitated as ThO. Precipitation was greater with electrolytic dissolution. The results observed in these studies suggest a dissolution cycle that permits separation of the canned-slug components. Stoichiometric use of the acid might be more closely approached and problems of materials of construction could be lessened. (auth)

7410 CF-51-8-120
Oak Ridge National Lab., Tenn.
DRY PROCESSING. I. CHLORINATION. II. HYDROFLUORINATION. Quarterly Report for Period May 10 to August 10, 1951. B. L. Baker. August 10, 1951. Decl. Feb. 13, 1957. 9p. Contract [W-7405-eng-26]. $1.80 (ph OTS); $1.80(mf OTS).

An exploratory study of dissolution by chlorination of metals and alloys of interest to AEC was made. High rates of reaction have been attained at temperatures up to 800°C with U–Al, U–Zr, and stainless alloys. Recovery of U and V from low-grade, high-silica, U ores by chlorination was partially investigated. Best results were 71% U and 81.5% V recoveries. Alundum was found to be highly resistant to chlorine at 800 to 900°C. Reaction rates between anhydrous HF and U–Zr alloy at 600°C are very low. (auth)

7411 CF-51-12-16
[Oak Ridge National Lab., Tenn.]
DESIGN CALCULATIONS FOR THORIUM DISSOLVER GAS COOLING AND SCRUBBING EQUIPMENT. R. J. Klotzbach. Dec. 3, 1951. Decl. Feb. 15, 1957. 24p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Design calculations for a dissolver reflux condenser for handling thorium slugs and a sodium hydroxide scrubbing tower for the removal of NO_2 from the dissolver off-gas are given. (F.S.)

7412 CF-53-8-149
Oak Ridge National Lab., Tenn.
THE ABRASIVE PROPERTIES OF THORIUM OXIDE. J. P. McBride. Aug. 21, 1953. Decl. Feb. 14, 1957. 4p. Contract [W-7405-eng-26]. $0.20(OTS).

Thorium hydroxide which has not been allowed to dry, after being precipitated and washed thoroughly, is completely non-abrasive even after autoclaving in H_2O at 250°C for as long as 64 hr. Thoria produced by ignition of $Th(C_2O_4)_2$ at 500°C for 120 hr and then heated as an aqueous slurry for 4 days at 250°C is less abrasive than a slurry of an oxide, prepared from $Th(C_2O_4)_2$ at ≥ 650°C, which has been subjected to a similar hydrolytic process. (auth)

7413 CF-54-7-97
Oak Ridge National Lab., Tenn.
PREPARATION OF ACID DEFICIENT ALUMINUM NITRATE BY ELECTRODIALYSIS. E. J. Parsi. July 21, 1954. Decl. Feb. 14, 1957. 20p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

The electrodialysis method for the preparation of acid-deficient $Al(NO_3)_3$ was tested in a continuous 100 hour run. Operating data, flowsheets, and cost estimates are given. (auth)

7414 CF-55-5-198
Oak Ridge National Lab., Tenn.
ThO_2 SLURRY DEVELOPMENT: QUARTERLY REPORT FOR PERIOD ENDING APRIL 30, 1955. D. E. Ferguson and J. P. McBride. May 20, 1955. Decl. Feb. 16, 1957. 36p. Contract [W-7405-eng-26]. $0.40(OTS).

The effects of in-pile irradiation on the properties of ThO_2 slurries were investigated. The development of an internal catalyst to effect the recombination of radiolytic gases produced in the slurry blanket was continued. The surface area, particle size, and thermogravimetric properties of ThO_2 prepared by the thermal decomposition of oxalate precipitated from Thorex product solution were determined. The viscosities of several ThO_2 slurries were measured. (C.W.H.)

7415 CF-55-6-71
Oak Ridge National Lab., Tenn.
COMMENTS ON THE EFFECT OF ELECTROLYTES ON THORIUM OXIDE SLURRIES. D. G. Thomas. June 8, 1955. Decl. Feb. 14, 1957. 13p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Since the physical properties of thorium oxide slurries are very sensitive to the species and quantity of electrolyte present, a study of the action of electrolytes was proposed. A brief literature survey is included. (B.J.H.)

7416 CF-55-8-130
Oak Ridge National Lab., Tenn.
PREPARATION OF THORIUM TETRACHLORIDE BY CHLORINATION OF THORIUM OXIDE OR THORIUM OXALATE. J. M. Chandler. Aug. 24, 1955. Decl. Feb. 14, 1957. 32p. Contract W-7405-eng-26. $6.30(ph OTS); $3.00(mf OTS).

An engineering evaluation is presented of the chlorination processes for $ThCl_4$ from Th oxides or oxalates. One-pound batches of 99% by weight $ThCl_4$ were prepared. Chlorination of the oxalates is the preferred procedure. (F.S.)

7417 CF-55-8-157
Oak Ridge National Lab., Tenn.
CHEMICAL TECHNOLOGY DIVISION, UNIT OPERATIONS SECTION MONTHLY PROGRESS REPORT [FOR] AUGUST 1955. W. K. Eister. Decl. Feb. 16, 1957. 70p. Contract [W-7405-eng-26]. $0.55(OTS).

7418 CK-1516
Ames Lab., Ames, Iowa.
CHEMICAL RESEARCH — CHEMISTRY OF PLUTONIUM REPORT FOR PERIOD OF FEBRUARY 1 TO MARCH 10, 1944. Decl. Feb. 16, 1957. 26p. Contract [W-7405-eng-82]. (A-2095). $4.80(ph OTS); $2.70(mf OTS).

Studies are reported on reduction of UF_6 to UF_4, conversion of Argonne U to UF_4, effect of various ions on LaF_3 separation of Pu, fluorination of UF_4 from Argonne metal, distillation of Pu in F_2, and formation of extractable complexes of Pu^{+4}. (L.M.T.)

7419 CN-2037
[Clinton Labs., Oak Ridge, Tenn.]
STUDIES IN THE PHYSICAL CHEMISTRY OF PRECIPITATE FORMATION: $BiPO_4$. Final Report [for] Period July 1, 1944 to January 1, 1945 [on] Problem Assignment No. 203-X72C. Part I. A. R. Brosi. June 1, 1945. Decl. Feb. 16, 1957. 75p. Contract W-7405-eng-39. $12.30(ph OTS); $4.50(mf OTS).

Data on the effect of various process conditions on the specific surface of the precipitated $BiPO_4$ are presented. The measurements were made by gas adsorption methods, and in some cases, by ion adsorption and ion exchange methods. Electron photomicrographs are presented which show the shape, size and size distribution of crystals in $BiPO_4$ precipitates. A technique for measuring particle size distributions using radioactive tracers is described. Recent theories of precipitate formation are discussed and applied to the experimental data. Qualitative agreement between experimental results and theory is indicated. A study of the changes in hexagonal $BiPO_4$ precipitates on digestion is reported. The most important change on long digestion is the conversion from the hexagonal to the monoclinic form. Quantitative data on the rate of this conversion and a theory of the mechanism are presented. (auth)

7420 CN-2224
[Clinton Labs., Oak Ridge, Tenn.]
THE COPRECIPITATION OF PLUTONIUM(IV) WITH BISMUTH PHOSPHATE. Final Report [for] Period September 15, 1944 to January 1, 1945 [on] Problem Assignment No. 203-X72C. Part III. Arthur W. Adamson. Jan. 15, 1945. Decl. Feb. 18, 1957. 30p. Contract W-7405-eng-39. $4.80(ph OTS); $2.70(mf OTS).

The co-precipitation of Pu^{4+} with α $BiPO_4$ at 75°C. obeys the equations for isomorphous carrying. With a fast, direct strike, the Doerner-Hoskins logarithmic equation was obeyed with $\lambda = 10$, whereas, under the same strike conditions, but keeping the Pu : Bi ratio in solution constant, quasi-Nernst behavior was found, with $K = 10$. With slow precipitation, long digestion, and with constant Pu : Bi ratio in solution, an equilibrium value of the Nernst distribution coefficient of $K = 120$ was obtained. It is concluded that to obtain maximum carrying, slow and complete precipitation under Doerner-Hoskins conditions is necessary, and that prolonged digestion as well as Pu complexing Agents are to be avoided. To explain the apparent isomorphous behavior of the two dissimilar phosphates, it is postulated that strong internal adsorption due to two dimensional lattice similarities takes place. (auth)

7421 CT-3045
[Chicago. Univ. Metallurgical Lab.]
A STUDY OF THE REACTION RATE BETWEEN THORIUM AND PURIFIED HYDROGEN. R. P. Straetz and J. E. Draley. June 11, 1945. Decl. Feb. 18, 1957. 7p. Contract W-7401-eng-37. $1.80(ph OTS); $1.80(mf OTS).

7422 FMPC-165(Rev. 1)
National Lead Co. of Ohio, Cincinnati.
STANDARD OPERATING PROCEDURE FOR ZINC CHLORIDE PROCESSING FOR USE IN THORIUM PRODUCTION (Section 2.5.3.9. First Revision). E. W. Mautz, O. R. Magoteaux, P. Kleinsmith, and W. B. Clymer. Jan. 25, 1954. Decl. Feb. 16, 1957. 7p. Contract AT(30-1)-1156. $0.25(OTS).

A step-by-step procedure is given for drying, weighing, and packaging $ZnCl_2$ for use in the reduction of ThF_4. (J.S.R.)

7423 FMPC-173
National Lead Co. of Ohio, Cincinnati.
REVIEW OF FMPC THORIUM PRODUCTION OPERATION AND SURVEY OF DEVELOPMENT NEEDS. E. W. Mautz, O. R. Magoteaux, and T. C. Runion. Mar. 26, 1953. Decl. Feb. 16, 1957. 24p. Contract AT-(30-1)-1156. $0.35 (OTS).

A survey of the Th production operation designed for the FMPC by the Catalytic Construction Company has been conducted to indicate the need for further development work in the event of full scale production requirements for Th, and to supply operating supervisors with a brief summary of the process chemistry and metallurgy relating to the standard operating procedures for the plant. (auth)

7424 FMPC-207
National Lead Co. of Ohio, Cincinnati.
OXIDATION OF CHLORIDES IN RAFFINATE MATERIALS. E. R. Johnson, L. M. Levy, R. L. Maddox, and J. Blum. Apr. 24, 1953. Decl. Feb. 16, 1957. 24p. Contract AT(30-1)-1156. $0.35(OTS).

The removal of chlorides from refining extraction raffinates has been investigated. The oxidants tested were Na_2O_2, H_2O_2, $Ce(SO_4)_2$, $K_2S_2O_8$, PbO_2, $KMnO_4$, $K_2Cr_2O_7$, and O_3, the most satisfactory being K_2MnO_4 and O_3. (J.S.R.)

7425 HW-17136
Hanford Works, Richland, Wash.
INFRARED ABSORPTION MEASUREMENTS INDICATING INTERMOLECULAR ASSOCIATION OF WATER AND HEXONE. R. H. Moore. March 8, 1950. Decl. Feb. 7, 1957. 10p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

Infrared absorption measurements in the region of the hydroxyl bands show that the hydroxyl bands of H_2O, dis-

solved in hexone or acetone, appear at lower frequency than those of H_2O dissolved in the inert solvents, CS_2 and CCl_4. This effect is taken as evidence for complex formation between the ketone and H_2O through the mechanism of the H bond. The carbonyl frequency of acetone shifts to lower frequencies upon the addition of H_2O, the shift increasing with increasing H_2O concentration. This effect is also attributed to complex formation involving a H bond. (auth)

7426 HW-20936
Hanford Works, Richland, Wash.
THE SOLUBILITY OF TRIBUTYL PHOSPHATE IN AQUEOUS SOLUTIONS. Leland L. Burger and R. C. Forsman. Apr. 2, 1951. Decl. Mar. 2, 1957. 17p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Solubilities of pure tributyl phosphate and of TBP in an inert diluent were determined in water and in solutions of nitric acid of various concentrations. Measurements were also made in solutions typical of various process streams and in more concentrated uranyl nitrate solutions. In the case of the TBP-diluent-water system the equilibrium water content in the organic phase was also determined. Data are tabulated. (C.H.)

7427 HW-29086
Hanford Atomic Products Operation, Richland, Wash.
PHYSICAL PROPERTIES OF TRIBUTYLPHOSPHATE-DILUENT SOLUTIONS. W. F. Johnson and R. L. Dillon. Sept. 1, 1953. Decl. Feb. 7, 1957. 25p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

In connection with the study of aqueous-tributylphosphate (TBP) systems, the study of the interactions of various diluents with the TBP has been extended. The diluents used were mineral oil, Amsco 125-90W, dodecane, n-hexane, cyclohexane, benzene, n-octyl bromide, carbon tetrachloride, chloroform, and pentachloroethane. The solubility of TBP in water from solutions of these diluents and the solubility of water in the organic phase was determined. For the paraffin hydrocarbons as diluents, the solubility of TBP in water was found to increase with increasing molal volume of the diluent. No evidence for the formation of a TBP water complex of simple mole ratio was found. The partial molal volume of TBP in all the diluents was constant up to a volume fraction of 0.75 and equal to the molal volume of TBP within ± 0.5 per cent. No simple relationship could be found which related the viscosity to the concentration for TBP in the diluents dodecane, n-hexane, cyclohexane, and pentachloroethane whether the solutions were dry or water-saturated. The self-diffusion coefficient of TBP was found to be 2.29×10^{-6} cm^2/sec. For dodecane and pentachloroethane solutions the product $D\eta$ was found to be essentially constant. (auth)

7428 ISC-186
Ames Lab., Ames, Iowa.
SEMI-ANNUAL REPORT IN CHEMICAL ENGINEERING FOR THE PERIOD APRIL 1, 1951–SEPTEMBER 30, 1951. G. L. Bridger, B. F. Ruth, G. H. Beyer and W. R. Millard. Nov. 18, 1951. Decl. Feb. 26, 1957. 23p. Contract W-7405-eng-82. $4.80(ph OTS); $2.70(mf OTS).

A progress report is presented covering the following research problems: extraction of Th from monazite, separation of Hf from Zr by adsorption, liquid-liquid extraction studies, and the separation of Hf and Zr by liquid-liquid extraction using tributyl phosphate–butyl ether solvent. (auth)

7429 NAA-SR-1476
Atomics International, North American Aviation, Inc., Canoga Park, Calif.
GENERAL CHEMISTRY QUARTERLY PROGRESS REPORT FOR APRIL–JUNE 1955. Eugene L. Colichman and E. E. Motta, eds. Mar. 15, 1956. Decl. Feb. 27, 1957. 22p. Contract AT-11-1-GEN-8. $0.25(OTS).

Additional results of polyphenyl irradiations employing 1-Mev electrons are reported. The suitability of tetralin as a reactor shield coolant is discussed. Results on a number of analytical chemistry development problems are presented. (auth)

7430 ORNL-336
Oak Ridge National Lab., Tenn.
CHEMISTRY DIVISION QUARTERLY REPORT FOR THE MONTHS DECEMBER 1948, JANUARY AND FEBRUARY 1949. May 4, 1949. Decl. Mar. 7, 1957. 123p. Contract W-7405-eng-26. $19.80(ph OTS); $6.30(mf OTS).

7431 UCRL-1664
California. Univ., Berkeley. Radiation Lab.
THE DISTRIBUTION OF Pu(VI) AND Pu(III) IN THENOYLTRIFLUOROACETONE–BENZENE–NITRIC ACID MIXTURES. D. L. Heisig and T. E. Hicks. Feb. 2, 1952. Decl. Feb. 8, 1956. 16p. Contract W-7405-eng-48. $3.30(ph OTS); $2.40(mf OTS).

The distribution coefficients of Pu^{6+} and Pu^{3+} were measured as a function of TTA, H^+, and NO_3^- concentrations in the TTA–C_6H_6–HNO_3 system. The extraction constants for Pu^{6+} and Pu^{3+} are 1.4×10^{-2} and 2×10^{-5}, respectively. (auth)

7432 Y-574
Oak Ridge National Lab., Y-12 Area, Oak Ridge, Tenn.
CHLORINATION OF PURIFIED ZIRCONIUM OXIDE. L. P. Twichell. Mar. 10, 1950. Decl. Feb. 7, 1957. 17p. Contract W-7405-Eng-26. $3.30(ph OTS); $2.40(mf OTS).

A series of experiments was made to study various methods of chlorination of ZrO_2. Several types of liquid phase reactions were tried without success. Chlorination of the oxide and added C by means of Cl_2 gas at high temperature gave incomplete conversion under the conditions studied. Reaction of oxide with CCl_4 vapor at 500°C gave essentially complete conversion to product of required purity. This was accomplished in a reactor and a condenser both made of nickel to simulate conditions in the large chlorinators available. The reaction was considerably more rapid at 550 to 600°C but the resultant product was highly contaminated with Ni. Studies were also made of the corrosion of Ni and other metals, types of condensers, and degree of ignition necessary in starting oxide. Cost estimates indicate that the oxide could be chlorinated by this method for $0.51 per pound of contained Zr metal. (auth)

ANALYTICAL PROCEDURES

7433 A-1267
Columbia Univ., New York. Div. of War Research.
ANALYSIS OF BORON TRICHLORIDE. Charles M. Judson. May 1, 1944. Decl. Feb. 14, 1957. 9p. Contract W-7405-eng-50. (100B-R-132). $1.80(ph OTS); $1.80(mf OTS).

Procedures are presented for the analysis of BCl_3 and the determination of $SiCl_4$ impurities. (C.W.H.)

7434 A-1299
Columbia Univ., New York. Div. of War Research.
ANALYSIS OF BORON FLUORIDE – METHYL ETHER COMPLEX. Charles M. Judson and Clarence D. Turner. July 27, 1944. Decl. Feb. 13, 1957. 13p. Contract W-7405-Eng-50. (100B-R-188). $3.30(ph OTS); $2.40(mf OTS).

A method for the determination of boron and fluorine in the methyl ether complex of boron fluoride is described. The procedure is a modification of that used by the

Harshaw Company for the analysis of boron fluoride gas. Experiments which led to the various modifications introduced are described. The method used is more accurate than those used by previous investigators for the analysis of boron fluoride complexes. Possible applications to other compounds of boron fluoride are discussed. The problem of a complete analysis of all the constituents of the methyl ether complex is considered. (auth)

7435 BMI-260
Battelle Memorial Inst., Columbus, Ohio.
THE DETERMINATION OF RARE EARTHS IN THORIUM. E. J. Center, W. M. Henry, and R. D. Householder. May 15, 1952. Decl. Feb. 14, 1957. 13p. Contract AT-(30-1)-Gen-228. $0.30(OTS).

A quantitative method for the determination of individual rare earths in Th down to a level of 0.05 ppm is described. The procedure consists of a chromatographic cellulose-column separation followed by a solution-type spectrographic determination. Values are given for the recovery of a number of rare earths using this combined procedure. (auth)

7436 CF-53-6-166
Oak Ridge National Lab., Tenn.
DIBUTYL AND MONBUTYL PHOSPHATE STUDIES: THE DETERMINATION OF MICRO-QUANTITIES OF DIBUTYL PHOSPHATE IN TRIBUTYL PHOSPHATE-AMSCO SOLUTIONS. C. E. Pietri. June 23, 1953. Decl. Feb. 14, 1957. 15p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The fluorophotometric determination of monobutyl and dibutyl phosphates in TBP-kerosene systems is described. The procedure is based on complex formation of the phosphates with U, and subsequent determination of the amount of U complexed. (C.W.H.)

7437 CF-55-6-27
Oak Ridge National Lab., Tenn.
EXAMINATION OF IRRADIATED ThO_2 SLUGS FOR THE U^{233}/Th RATIO. R. P. Wischow. June 2, 1955. Decl. Feb. 14, 1957. 6p. Contract [W-7405-eng-26]. $1.80 (ph OTS); $1.80(mf OTS).

7438 HW-15944
Hanford Works, Richland, Wash.
THE DETERMINATION OF TOTAL PLUTONIUM IN THE PRESENCE OF ALUMINUM. R. E. Burns and G. B. Barton. Feb. 10, 1950. Decl. Feb. 19, 1957. 7p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

The adoption of aluminum nitrate as salting agent in the Redox process made it imperative that a method be available for determining plutonium in the presence of aluminum. However, large amounts of aluminum have been found to interfere with the determination of plutonium by the lanthanum fluoride procedure. Previous attempts to increase the accuracy of the lanthanum fluoride method, by precipitating LaF_3 from 4M HF (rather than 2M), have been successful only when the initial plutonium level was high. A method has been developed which permits greater than 96% plutonium recoveries from 1.5M aluminum nitrate solutions. Quantitative carrying of plutonium (mixed oxidation states) on a lanthanum hydroxide–sodium diuranate precipitate serves to separate the plutonium from the aluminum as well as from any dichromate present. The precipitate, following washing with dilute caustic, is dissolved in 2M HCl. A standard lanthanum fluoride procedure then follows. An outline of the complete procedure is included. (auth)

7439 HW-17543
Hanford Works, Richland, Wash.
THE DETERMINATION OF URANYL NITRATE IN PRETREATED HEXONE BY INFRARED ABSORPTION MEASUREMENTS. R. H. Moore. Apr. 20, 1950. Decl. Feb. 20, 1957. 10p. Contract W-31-109-Eng-52. $1.80 (ph OTS); $1.80(mf OTS).

The determination of uranium nitrate hexahydrate in pretreated hexone was studied in the range 0 to 40 g/l. The analysis can be performed with a precision of ± 1.28 g/l. (auth)

7440 KLX-1063
Kellex Corp., New York.
DEVELOPMENT OF MACRO AND MICRO ANALYTICAL PROCEDURES FOR PROCESS SOLUTIONS. Summary Report covering period July 18, 1949 to Feb. 17, 1950. Mar. 7, 1950. Decl. Feb. 25, 1957. 14p. $3.30(ph OTS); $2.40(mf OTS).

The composition and empirical formulae of the soda salt precipitates obtained by treatment of simulated current metal waste with caustic soda were determined by chemical analysis. Procedures are presented for the determination of Na in process solutions and for the chemical separation of Sr from Y. (C.W.H.)

7441 NAA-SR-237
North American Aviation, Inc., Downey, Calif.
THE DETERMINATION OF URANIUM IN A MIXTURE OF BROMINE, BROMINE TRIFLUORIDE AND URANIUM HEXAFLUORIDE. W. J. Gardner. May 13, 1953. Decl. Mar. 2, 1957. 10p. Contract AT-11-1-gen-8. $1.80(ph OTS); $1.80(mf OTS).

A method of analysis for U in a mixture of Br, BrF_3 and UF_6 is described. The UF_6 is distilled over along with some Br_2 and BrF_3 into a bed of crystalline H_3BO_3, where it is absorbed. The H_3BO_3 reaction mixture is heated after the addition of H_2SO_4 and excess HF. Volatile F compounds and B compounds are driven off to leave a dry residue of UO_2SO_4. U in the UO_2SO_4 is determined by standard methods. (auth)

FLUORINE AND FLUORINE COMPOUNDS

7442 A-2121
Columbia Univ., New York. Div. of War Research.
CONVERSION OF $(CH_3)_2O \cdot BF_3$ TO BF_3. Abraham D. Kirshenbaum and Clyde A. Hutchison. Nov. 30, 1944. Decl. Feb. 12, 1957. 81p. Contract W-7405-eng-50. (100B-R-259). $24.30(ph OTS); $7.50(mf OTS).

A technique for the conversion of $(CH_3)_2O \cdot BF_3$ to BF_3 is outlined. Equations summarizing the two-step process are: $(CH_3)_2O \cdot BF_3 + x\ CaF_2 \rightleftharpoons (CH_3)_2O + x\ CaF_2 \cdot BF_3$ and $x\ CaF_2 \cdot BF_3 \rightleftharpoons CaF_2 + BF_3$. (C.W.H.)

7443 A-2155
Columbia Univ., New York. Substitute Alloy Materials Lab.
CONSTITUENT ANALYSIS OF BORON FLUORIDE–METHYL ETHER COMPLEX. Charles M. Judson. Nov. 15, 1944. Decl. Feb. 12, 1957. 9p. Contract W-7401-eng-50. (100B-R-241). $1.80(ph OTS); $1.80(mf OTS).

Methods for determining SiF_4, Fe, Ni, Cu, and H_2O in the boron fluoride–methyl ether complex are outlined. (D.F.)

7444 A-2355
Standard Oil Co. of Indiana, Chicago.
DISSOCIATION OF THE METHYL BORATE-BORON FLUORIDE COMPLEX. D. A. McCaulay. Sept. 11, 1945. Decl. Feb. 12, 1957. 14p. Contract W-7418-eng-41. $3.30(ph OTS); $2.40(mf OTS).

By combining data previously obtained in this laboratory with data available in the literature, it was found possible to calculate the degree of dissociation and the heat of vaporization of methyl borate–boron fluoride complex. The degree of dissociation at the saturation pressure was calculated to be less than 1%. (auth)

GRAPHITE

7445 BNL-1896
Brookhaven National Lab., [illegible]on, N. Y.
ADDITIONAL EXPERIMENT[illegible]ON STORED ENERGY IN BNL REACTOR GRAPHITE. [illegible]. Mulhern. Aug. 3, 1953. Decl. Feb. 13, 1957. 25p. Contract [AT-30-2-gen-16]. $4.80(ph OTS); $2.70(mf OTS).

Time-temperature curves for normal and irradiated graphite were obtained from the same sample by heating an irradiated sample to release the stored energy and reheating after the energy was released. (K.S.)

7446 CP-2242
Chicago. Univ. Metallurgical Lab.
THE PENETRATION OF GRAPHITES OF DIFFERENT PURITIES BY LEAD BISMUTH EUTECTIC. Final Report. William G. Stroud. Sept. 25, 1944. Decl. Feb. 18, 1957. 15p. $3.30(ph OTS); $2.40(mf OTS).

Bi–Pb eutectic and pure Bi penetration of graphites of various densities under 300°C and pressures up to 90 psi was found to be of the order of magnitude of 1.0 g/cm^2 and was uniform throughout the graphite. Bi was absorbed less than the eutectic but the oxidation rate of Bi would be prohibitive. There was heavy nitriding in N atmospheres with the Bi and the eutectic. (F.S.)

7447 HW-8767
Hanford Engineer Works. GE Nucleonics Project, Richland, Wash.
CORRELATION OF CHEMICAL ANALYSES ON GRAPHITE BARS AND TEST PILE RESULTS. J. J. Lane. Feb. 5, 1948. Decl. Feb. 19, 1957. 2p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

The chemical analyses of graphite test bars obtained from National Carbon Company and the analyses performed by the Hanford Works Laboratories Division did not correlate significantly with the test pile results, nor did the chemical analyses obtained from National Carbon Company agree with those done in the Hanford Works Laboratories. Chemical results obtained at Hanford on samples taken lengthwise from individual bars correlated significantly with the individual test pile results. (auth)

7448 HW-29082
Hanford Atomic Products Operation, Richland, Wash.
SURFACE STUDIES OF IRRADIATED GRAPHITE. C. N. Spalaris. July 14, 1953. Decl. Feb. 7, 1957. 28p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

The effects of oxidation and irradiation on the microsurface structure of pile graphite have been investigated by measuring the surface area and pore size distribution of several samples. The results obtained for both oxidized and irradiated graphite samples indicate that changes in surface characteristics which occur are determined by the flux, temperature of irradiation, and gaseous atmosphere in which the radiation takes place. (auth)

7449 HW-30719
Hanford Atomic Products Operation, Richland, Wash.
ARTIFICIAL GRAPHITE AS A NUCLEAR REACTOR MATERIAL. L. P. Bupp. Feb. 1, 1954. Decl. Feb. 19, 1957. 19p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

Data concerning the behavior of artificial graphite as a nuclear reactor material are summarized. Included are brief discussions on the structure of graphite and the effect of radiation on various properties of graphite. (B.J.H.)

7450 NAA-SR-41
North American Aviation, Inc., Bowney, Calif.
ON THE MECHANISM OF THERMAL ANNEALING OF IRRADIATED GRAPHITE. Dwain Bowen. Aug. 4, 1950. Decl. Feb. 7, 1957. 38p. Contract AT-11-1-GEN-8. $6.30(ph OTS); $3.00(mf OTS).

Reaction rate theory has been applied to the thermal annealing of radiation damage in graphite. The property examined in detail is electric resistivity because the connection between lattice disturbance and electric resistivity has been theoretically studied. In order to achieve maximum consistency of experimental data and theoretical analysis, it is necessary to assume that the state removed between 100 and 800°C corresponds to a sixth order reaction, the state removed between 900 and 1300°C corresponds to a first order reaction, and that the state removed between 1300° and 2000°C corresponds to a fourth order reaction. If it is accepted that the order of a reaction indicates the number of microscopic elements that must cooperate, then the first state may be characterized by the formation of clusters of six interstitial atoms (perhaps as a benzene ring structure). It has been shown that these rings are probably located at the crystalite boundry. This implies that the single interstitial atoms may diffuse between the planes without being trapped until the clusters of six atoms form at the boundary. The first order reaction agrees with the theoretical assumption that the second state corresponds to the annealing of lattice vacancies. The fourth order reaction of the third state may be interpreted to indicate that four clusters of the type described above unite, either by joining the adjacent crystallite, or by forming new microcrystals. (auth)

Refer also to abstract 7904.

LABORATORIES AND EQUIPMENT

7451 CF-52-5-202
Oak Ridge National Lab., Tenn.
PULSE GENERATOR DEVELOPMENT. Quarterly Report for Period February 10, 1952 to May 10, 1952. J. E. Conaway and A. C. Jealous. May 26, 1952. Decl. Feb. 13, 1957. 14p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The effects of piston construction in the piston-type pulse generator and the improvement of pulse generator drive systems are discussed. (B.J H.)

7452 CF-52-6-107
[Oak Ridge National Lab., Tenn.]
AN EXPERIMENTAL COMPARISON OF THE HIGGINS CONTINUOUS ION EXCHANGER WITH A CONVENTIONAL FIXED-BED ION EXCHANGE COLUMN. J. T. Roberts. June 18, 1952. Decl. Feb. 14, 1957. 18p. $3.30(ph OTS); $2.40(mf OTS).

Under conditions of equal flow rates, flow ratios, and resin bed sizes a laboratory-scale, hand-operated model of a continuous-countercurrent ion exchanger invented by I. R. Higgins gave better results than a conventional fixed-bed ion exchange column in typical recovery-concentration and separation processes. (auth)

7453 KAPL-609
Knolls Atomic Power Lab., Schenectady, N. Y.
AN IMPROVED MINIATURE MIXER-SETTLER. B. L. Zebroski, J. W. Codding, B. V. Coplan, and H. W. Alter. Sept. 10, 1951. Decl. Feb. 7, 1957. 21p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

The development of an improved model of the miniature mixer-settler is reported. Results with three different liquid-liquid extraction systems are given for a twelve-stage unit. Simple and reliable hydraulic operation is routinely obtained. The unit has mixing chambers 3/4 inch square in cross section. Conveniently operable throughputs range from about 37 to 450 cc/hr; flooding occurs at some-

thing over 550 cc/hr in a hexone–aluminum nitrate system. It was shown that the best operation was obtained with all impellers rotating in the same direction relative to the exit ports. Stage efficiencies of 73% to 92% have been found. Construction details of the unit are given. (auth)

7454 MonT-205
Clinton Labs., Oak Ridge, Tenn.
EXPERIENCE IN EQUIPMENT AND BUILDING DECONTAMINATION IN THE MANUFACTURE OF Ba^{140}. E. J. Witkowski. Nov. 19, 1946. Decl. Mar. 7, 1957. 7p. Contract W-35-058-Eng-71. $1.80(ph OTS); $1.80(mf OTS)

Solutions to problems encountered in decontamination of buildings and equipment are reported. General applications of some of the techniques and methods may be possible. (T.R.H.)

RADIATION CHEMISTRY

7455 ANL-4833
Argonne National Lab., Lemont, Ill.
SUMMARY REPORT FOR JANUARY, FEBRUARY, AND MARCH 1952; CHEMISTRY DIVISION, SECTION C-II. J. R. Gilbreath and O. C. Simpson, comps. Oct. 15, 1952. Decl. Feb. 12, 1957. 118p. Contract W-31-109-eng-38. $18.30(ph OTS); $6.00(mf OTS).

Progress is reported on the following studies: physical properties of graphite, radiation effects on the properties of graphite, radiation effects on physical properties of diamond, silica, and other ceramics, radiation effects on ice, color centers and other optical properties of single crystals, evaluation of an integral occurring in connection with the absorption of radiation in cylinders, radiation chemistry of liquids, vapor pressure and heat of vaporization of U, nuclear properties of Zr^{93} and Nb^{93} from fission, slow-neutron absorption cross section of graphite, radiations of Bi^{213}, Pb^{209}, Tl^{209}, and Ra^{225}, a graphical criterion for adding curves, proposal for a lens-type pair spectrometer, and paramagnetic resonance phenomena at radio frequencies. (K.S.)

7456 CC-1906
[Chicago. Univ.] Metallurgical Lab.
CHEMICAL RESEARCH–RADIATION CHEMISTRY REPORT FOR MONTH ENDING JUNE 30, 1944. Decl. Feb. 16, 1957. 18p. Contract W-7401-eng-37. (A-2695). $3.30(ph OTS); $2.40(mf OTS).

7457 CC-1969
[Chicago. Univ. Metallurgical Lab.]
CHEMICAL RESEARCH–RADIATION CHEMISTRY REPORT FOR PERIOD ENDING JANUARY 15, 1945. Decl. Feb. 16, 1957 8p. Contract W-7401-eng-37. (A-3031). $1.80(ph OTS); $1.80(mf OTS).

RADIATION EFFECTS

7458 CF-53-4-277
Oak Ridge National Lab., Tenn.
RECOMBINATION AND REACTOR-IRRADIATION BEHAVIOR OF URANYL PHOSPHATE–PHOSPHORIC ACID SOLUTIONS. R. M. Bidwell and H. F. McDuffie. Apr. 29, 1953. Decl. Feb. 14, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

7459 HW-13282
[Hanford Works, Richland, Wash.]
PARIAN AND TEFLON IN 1A REDOX SOLUTIONS. W. W. Koenig. May 9, 1949. Decl. Feb. 19, 1957. 3p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Data from 10-day exposure tests of these two materials are presented. Data indicate that each may be employed with specific solutions. (L.M.T.)

RARE EARTHS AND RARE-EARTH COMPOUNDS

7460 CN-2558
[Clinton Labs., Oak Ridge, Tenn.]
THE COPRECIPITATION OF THE TRIVALENT RARE EARTHS WITH BISMUTH PHOSPHATE. Final Report for the Period June 15, 1944 to January 1, 1945 on Problem Assignment No. 203-X72C. Part IV. L. S. Meyers, Jr. Jan. 15, 1945. Decl. Feb. 16, 1957. 48p. Contract W-7405-eng-39. $7.80(ph OTS); $3.30(mf OTS).

The Berthelot-Nernst distribution constant for the incorporation of La by α-$BiPO_4$ is $5 \pm 1 \times 10^{-4}$ at 25° and $2 \pm 1 \times 10^{-3}$ at 75°C. It was concluded that La is carried by isomorphous coprecipitation. At 25°C the incorporation consisted of a rapid pickup due to surface adsorption and a slower pickup due to recrystallization. (auth)

SEPARATION PROCEDURES

7461 A-2356
[Standard Oil Co. of Indiana, Chicago]
OPERATION OF A PODBIELNIAK HELIGRID COLUMN. W. L. Rittschof. Oct. 2, 1945. Decl. Feb. 12, 1957. 21p. Contract W-7418-eng-41. $3.30(ph OTS); $2.40(mf OTS).

The need for small quantities of concentrated polymer prior to the startup of the fractionation unit at this location made it desirable to undertake the preparation of this material in a laboratory Heligrid column. The distillation was run "upside down" since the operation was one of stripping out a small amount of desired "heavy" constituent from a mixture containing a large volume of light constituent. The method of operation proved successful with the result that the desired material was obtained in time to prepare standards for the spectrometer, and to obtain analytical experience before operation of the large scale unit was started. It was also demonstrated that no complicating factors should be encountered in the process due to accumulation of impurities in the higher concentration ranges. Performance data on the column indicated the presence of 80 to 150 theoretical plates, which corresponds to a height of 0.5 to 0.9 inch per theoretical plate. These results indicate Heligrid packing to be far superior to any other laboratory packing investigated for this separation, and strongly suggest its application to other laboratory fractionation problems of this nature. (auth)

7462 A-2359
[Standard Oil Co. of Indiana, Whiting.]
ANALYSIS OF DIMETHYL ETHER-BORON TRIFLUORIDE. Charles M. Judson. Oct. 26, 1945. Decl. Feb. 12, 1957. 50p. Contract W-7418-eng-41. $10.80(ph OTS); $3.90(mf OTS).

The methods used for laboratory control analyses in the operation of a unit for the separation of boron isotopes by the fractionation of dimethyl ether–boron trifluoride complex are described and evaluated. Procedures developed at the S.A.M. Laboratories of Columbia University for the determination of boron, fluorine, and dimethyl ether in this complex have been applied with only minor modifications. Procedures for the determination of impurities in the complex have been developed. Hydroxyl and alkoxyl groups are determined by methods employing the Karl Fischer reagent. Metals are determined by the conventional methods of colorimetric, volumetric, and gravimetric analysis. Conductivity measurements are also used in analytical control. Consideration has been given

to other impurities which might be expected to appear in the complex. Data have been presented to show the precision and accuracy of the methods. Consideration has been given to the chemical information obtainable from the analyses. Most of the methods are capable of further improvement. A great deal of further information is required before the chemical reactions of the impurities in dimethyl ether–boron trifluoride complex are completely understood. Some suggestions have been made as to the directions which further investigations might take. (auth)

7463 A-2370
Standard Oil Co. of Ind., Chicago.
CONSTRUCTION, OPERATION, AND MAINTENANCE OF A POLYMER FRACTIONATION UNIT. J. R. Eichna, K. G. Carpenter, C. J. McManus, R. Tuck, and C. Risher. July 23, 1946. Decl. Feb. 12, 1957. 105p. Contract W-7418-eng-41. $16.80(ph OTS); $5.70(mf OTS).

The vacuum fractionation unit was operated for the production of heavy polymer for a period of over a year. Experience gained during the operation of the unit indicates that a number of precautions should be taken in the construction of a similar unit. It is of utmost importance that the entire system be thoroughly tested for leaks. In view of the fact that small leaks tend to become enlarged during operation, it was necessary to use special techniques to find these leaks. It was recommended that all items of equipment such as heat exchangers, column sections, and piping be individually tested and proved to be completely free of leaks before any attempt is made to operate the unit. Flow meters which do not employ mercury were recommended for use in polymer lines. Stedman packing or other packings having equivalent characteristics should be considered in any future installation. To obtain satisfactory performance of the packing, it is important that efficiency tests be made prior to installation, and that liquid redistributors be used at intervals of not less than five feet. Water is suitable as a cooling medium and steam as a heating medium provided that careful attention has been paid to the elimination of exchanger leaks. Steam ejectors are recommended for use in providing the vacuum for the unit, but it is highly advisable that spare vacuum facilities be installed to ensure continuity of operation. (auth)

7464 A-3789(Sec. 7-A)
Du Pont de Nemours (E. I.) and Co. TNX Div., Wilmington, Del.
CLINTON LABORATORIES PROCESS MANUAL. SECTION 7-A. PRODUCT PURIFICATION WET D PROCESS. W. E. Kirst. Sept. 27, 1943. Decl. Feb. 12, 1957. 34p. $6.30(ph OTS); $3.00(mf OTS).

7465 ANL-4019
Argonne National Lab., Lemont, Ill.
THE EXTRACTION OF IONIUM BY ORGANIC SOLVENTS. Earl K. Hyde and Jean Tolmach. Aug. 11, 1947. Decl. Feb. 12, 1957. 20p. Contract W-31-109-eng-38. $3.30 (ph OTS); $2.40(mf OTS).

A 6N calcium nitrate solution acidified to 2N with HNO_3, and an 8N ammonium nitrate solution acidified to 2N with HNO_3 were spiked with Th^{230}, and different solvents were used for extracting the Th^{230}. Diethyl cellosolve extraction was high, but its physical properties forbid its use in a column. Pentaether was found to be an excellent solvent and quite suitable for batch operations. (F.S.)

7466 ANL-4382
Argonne National Lab., Lemont, Ill.
SEPARATION OF NEPTUNIUM AND PLUTONIUM CHLORIDES BY A FLOW DISTILLATION METHOD. John G. Malm. Oct. 1949. Decl. Feb. 12, 1957. 16p. Contract W-31-109-eng-38. $0.30(OTS).

Neptunium was separated from Pu by chloride formation and subsequent distillation in a N_2 atmosphere at 500 to 600°C. The Pu concentration in the $NpCl_4$ product was ~1 ppm. (C.W.H.)

7467 ANL-4442
Argonne National Lab., Lemont, Ill.
RECOVERY OF NEPTUNIUM-237 FROM SPECIAL HANFORD WASTES. H. B. Evans, W. B. Seefeldt, and H. H. Hyman. Mar. 27, 1950. Decl. Feb. 12, 1957. 26p. Contract W-31-109-eng-38. $4.80(ph OTS); $2.70(mf OTS).

Neptunium-237 was recovered with moderate success from eight batches of special Hanford Pu waste solutions by a batch solvent extraction process. The procedure utilized the behavior of Np-IV to form highly solvent-soluble nitrate complexes in the presence of certain organic amines or quarternary salts under conditions where Pu-III can be stabilized. Some very preliminary work on an alternate batch solvent extraction process utilizing the solvent inextractability of Np-V is also reported. The feasibility of continuous countercurrent extraction for the separation of Np from Pu is discussed. (auth)

7468 BMI-242
Battelle Memorial Inst., Columbus, Ohio.
ANION-EXCHANGE STUDIES ON THE SEPARATION AND PURIFICATION OF URANIUM AND THORIUM. R. H. Poirier and A. E. Bearse. Aug. 15, 1950. Decl. Feb. 14, 1957. 71p. Contract AT-30-1-Gen-228. $0.55(OTS).

The separation of U from Th in carbonate, HCl, and H_2SO_4 solutions by anion-exchange resins is described. Capacities, influence of extraneous ions, and elution characteristics are discussed for various types of resins. The removal of U and Th from simulated process solutions, similar to those encountered in processing monazite sand, by the application of anion-exchange resins is described. The U–Th separation by selective adsorption of U on anion exchangers was very effective in Na_2CO_3 and HCl solutions, but only a partial separation was obtained in H_2SO_4 solutions. (auth)

7469 BMI-243
Battelle Memorial Inst., Columbus, Ohio.
RECOVERY OF THORIUM AND URANIUM FROM MONAZITE SAND. VOLUME I. Final Report. G. D. Calkins, R. B. Filbert, Jr., A. E. Bearse, and J. W. Clegg. Sept. 15, 1950. Decl. Feb. 14, 1957. 49p. Contract AT-30-1-Gen-228. $0.40(OTS).

A process for recovering high yields of U and Th from monazite sand concentrates is described. The monazite is decomposed by treatment with aqueous NaOH and the resulting hydrous metal oxides are dissolved in a mineral acid. A separation from rare earths is accomplished by partial neutralization of the acid solution. The resulting precipitate containing Th and U is dissolved in HNO_3. The Th and U are subsequently recovered and separated by solvent-extraction techniques using TBP. An alternative separation technique involving ion exchange was also evaluated. By-products of the monazite process are rare earths, in a high state of purity, and Na_3PO_4. (auth)

7470 BMI-246
Battelle Memorial Inst., Columbus, Ohio.
REFINING OF MgX AND OTHER URANIUM–BEARING MATERIALS. Progress Report [for] May 1, 1950 to August 31, 1950. R. A. Ewing, S. J. Kiehl, Jr., and A. E. Bearse. Aug. 31, 1950. Decl. Feb. 21, 1957. 77p. Contract AT-30-1-Gen-228. $0.55(OTS).

Results of two methods of reducing the phosphate and vanadate contents of MgX, a uranium precipitate from the Belgian Congo, are described. Leaching with caustic soda solution was slightly more effective than selective precipi-

tation by controlled neutralization of a nitric acid solution. The adverse effects of sulfate on the extraction of uranium by diethyl ether were examined in synthetic and process solution. Iron appeared to be an effective complexing agent for sulfate. The solubility of two uranium-vanadium compounds in synthetic process solutions was studied. Chemical analyses of several samples of pitchblende and MgX are presented. Attempts to identify the phases in MgX by x-ray and electron-diffraction techniques were unsuccessful. (auth)

7471 BMI-247
Battelle Memorial Inst., Columbus, Ohio.
REFINING OF MgX AND OTHER URANIUM-BEARING MATERIALS. Progress Report [for] September [1950]. R. A. Ewing, R. L. Belcher, S. J. Kiehl, Jr., and A. E. Bearse. Sept. 30, 1950. Decl. Feb. 21, 1957. 25p. Contract AT-30-1-gen-228. $0.35(OTS).

The effects of solution concentration and acidity on extraction of uranium by diethyl ether have been investigated. Data on the transference of vanadium and molybdenum to the ether phase were obtained in some tests. Continuous extractions were made with a Scheibel one-inch laboratory column at feed rates of 500 to 3000 ml./hr., and stirrer speeds of 1800 to 3400 r.p.m. The approximate solubility of vanadium pentoxide in nitric acid was determined. (auth)

7472 BMI-248
Battelle Memorial Inst., Columbus, Ohio.
REFINING OF MgX AND OTHER URANIUM-BEARING MATERIALS. Progress Report. R. A. Ewing, R. L. Belcher, S. J. Kiehl, Jr., and A. E. Bearse. Oct. 31, 1950. Decl. Feb. 21, 1957. 25p. Contract AT-30-1-Gen-228. $0.35(OTS).

The effects of acidity on extraction of V and Mo by diethyl ether are described. Solubility data for several vanadium compounds in nitrate process solutions are presented. Preliminary results on the extraction of U from process slurries by tributyl phosphate are reported, including data on the contaminants extracted with the U. (auth)

7473 BMI-249
Battelle Memorial Inst., Columbus, Ohio.
REFINING OF MgX AND OTHER URANIUM-BEARING MATERIALS. Progress Report. R. A. Ewing, R. L. Belcher, S. J. Kiehl, Jr., and A. E. Bearse. Nov. 30, 1950. Decl. Feb. 14, 1957. 18p. Contract AT-30-1-Gen-228. $0.30(OTS).

Results of tests investigating the effects of feed and scrub acidities on the extraction of U and contaminants from process slurries by tributyl phosphate-Gulfspray Naphtha in a Scheibel continuous extractor are described. Included are the results of tests on the effect of temperature on the reextraction of uranium from the organic to the aqueous phase. Studies on the fractional precipitation of MgX from a dilute sulfuric acid system throw considerable light on the probable constitution of MgX. Further data on the effect of acidity on the extraction of molybdenum and vanadium into ethyl ether are presented. (auth)

7474 BMI-256
Battelle Memorial Inst., Columbus, Ohio.
REFINING OF MgX AND OTHER URANIUM-BEARING MATERIALS. Quarterly Progress Report. R. A. Ewing, S. J. Kiehl, Jr., R. E. Sharpe, and A. E. Bearse. Sept. 30, 1951. Decl. Feb. 13, 1957. 42p. Contract AT-30-1-gen-228. $0.40(OTS).

Results are presented which suggest the possibility of satisfactorily extracting U from pitchblende feeds with neutral ether, without the necessity of removing sulfate, by the addition of a sufficient quantity of iron to the feed. The effect of adding minimal quantities of Fe on extraction with neutral ether was investigated for a number of MgX feeds, with and without V-20 soda salt added. The effect of temperature on extraction of U with a Scheibel column was studied and qualitatively evaluated. Arsenic was traced through both primary and secondary extraction stages in two tests; results indicate that As is satisfactorily rejected by a two-stage extraction, even from feeds relatively high in arsenic content. (auth)

7475 CF-49-1-239
Oak Ridge National Lab., Tenn.
TECHNICAL DIVISION–PILOT PLANT SECTION REPORT FOR MONTH ENDING JANUARY 15, 1949. D. G. Reid. Jan. 25, 1949. Decl. Feb. 14, 1957. 31p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

Five additional first cycle and six second cycle runs were completed in the Redox pilot plant. The first cycle decontamination factors obtained for plutonium and uranium under ANL flowsheet conditions were lower than the corresponding values obtained at ANL by factors of ten and five respectively. Hexone impurities and dissolver procedures were studied in an effort to increase zirconium and ruthenium decontamination. Plutonium losses in the IAW stream during the month were found to be consistently less than 0.01%, as determined by the TTA extraction method. In the second cycle the two acid deficient flowsheets developed at ORNL continued to give decontamination factors of about 300 and 900 respectively. A program is presented for the waste tank evaporator project. (auth)

7476 CF-51-4-176
Oak Ridge National Lab., Tenn.
DISTILLATION COLUMN FOR SEPARATING URANIUM AND FISSION PRODUCT FLUORIDES. J. M. Lenoir. Apr. 18, 1951. Decl. Feb. 16, 1957. 22p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

An all nickel distillation column, 11/16 in.-i.d., packed 8 inches deep with 5/32 Fenske helices has been designed and constructed. The column was built to operate at 75°C and 1600 mm of Hg pressure for the purpose of purifying U from its fission products by fractionation of metal fluorides. Remote control and protective shielding are features of the design. Operation of this column with fission product contaminated UF_6 should permit evaluation of relative volatilities, the degree of fluoride deterioration, and techniques of handling the distillation. From these data it is hoped that a pilot plant, consisting of two major columns and two small supplementary columns can be rationally designed for continuous production of UF_6. (auth)

7477 CF-51-9-115
Oak Ridge National Lab., Tenn.
PRESSURE DROP CALCULATIONS FOR THE OFF-GAS SYSTEM. H. O. Weeren. Sept. 24, 1951. Decl. Feb. 20, 1957. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Complete figures are tabulated for the pressure drop through the off-gas system for the 25 Process. (B.J.H.)

7478 CF-51-10-31
Oak Ridge National Lab., Tenn.
CORROSION OF TYPE 347 STAINLESS STEEL BY 1st CYCLE RAFFINATE. A. R. Olsen. Oct. 4, 1951. Decl. Feb. 20, 1957. 4p. Contract [W-7405-eng-26]. $1.80 (ph OTS); $1.80(mf OTS).

A boiling corrosive medium of pH 1.3 consisting of 2.2 M $Al(NO_3)_3$, $Hg(NO_3)_2$–5 wt.% of Al as Hg, and 0.79 M NH_4 (added as NH_4OH to produce 0.2 normal acid deficiency) was used for these tests. Tabular data are presented. (L.M.T.)

7479 CF-51-10-71
[American Cyanamid Co., New York.]
SPECIAL PROCESS EQUIPMENT. E. R. Edmondson.

Oct. 12, 1951. Decl. Feb. 13, 1957. 52p. $9.30(ph OTS); $3.60(mf OTS).

The report on the special process equipment includes, the solvent extraction columns, samplers, slug chargers, filters, agitators, pumps, jet syphons, valves, tube fittings, cranes, and service waste monitoring and metering equipment. (J.E.D.)

7480 CF-51-12-158
Oak Ridge National Lab., Tenn.
PUREX PROCESS FLOWSHEET NO. 3 IN TERMS OF PILOT PLANT FLOWS. D. O. Darby. Dec. 31, 1951. Decl. Feb. 15, 1957. 4p. $1.80(ph OTS); $1.80(mf OTS).

7481 CF-52-2-113
Oak Ridge National Lab., Tenn.
THE PRIMARY EXTRACTION MECHANISMS OF NITRIC ACID AND THORIUM NITRATE EXTRACTION BY TBP. Quarterly Report [for] Period November 10, 1951–February 10, 1952. Chemical Technology Div. J. E. Savolainen. Feb. 13, 1952. Decl. Feb. 16, 1957. 18p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Results obtained on the simultaneous extraction of HNO_3 and $Th(NO_3)_4$ from aqueous solutions by TBP in a paraffin hydrocarbon diluent indicate that $Th(NO_3)_4 \cdot 4TBP$ is the primary form in which Th is extracted. The extraction of HNO_3 by 10 and 12.5% TBP in hexane has been found to show inflections of the distribution coefficient curve in those regions where the HNO_3/TBP ratio approaches 0.25, 0.5, and 0.75 and probable inflections at 0.125 and 0.875. This behavior is not noticed on extraction with 7.5% TBP. It is postulated that with high $Th(NO_3)_4$ concentration in TBP a polymerization occurs, resulting in the formation of a compound having the limiting composition of $Th(NO_3)_4 \cdot 2TBP$. This postulate may account for TBP/Th ratios of less than 4 in the organic phase and also for the formation of two organic phases at high Th concentrations and the dispersing effect of an aromatic diluent on these two organic phase systems. (auth)

7482 CF-52-2-215
Oak Ridge National Lab., Tenn.
CONTINUOUS EVAPORATION OF THE PUREX PROCESS ICU STREAM IN THE ORNL PILOT PLANT. A. E. Wible. Feb. 27, 1952. Decl. Feb. 13, 1957. 12p. $3.30(ph OTS); $2.40(mf OTS).

Holdup in the ICU continuous evaporator is equivalent to about 1½-hr. production, and control is fully automatic. At 20% excess boil-up the evaporator recovers from sudden purges of feed in a reasonable time. Increasing the boildown required at the same U throughput has no effect on operation. A continuous chart record of product specific gravity at steady conditions showed fluctuations of ± 0.01 specific gravity units around a given set point. The equipment is described in detail. (J.E.D.)

7483 CF-52-4-13
Oak Ridge National Lab., Tenn.
PUREX 1A COLUMN STUDIES. J. R. Flanary. April 1, 1952. Decl. Feb. 20, 1957. 13p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The effects of aqueous-phase HNO_3 concentration and solvent saturation with U on the separation of fission products from Pu and U in the 1A column were investigated. Freezing point data of feed solutions and specific gravities of organic solvent phases containing U are reported. (C.W.H.)

7484 CF-52-5-95
Oak Ridge National Lab., Tenn.
THORIUM DISSOLVING–QUARTERLY REPORT [FOR] PERIOD–FEBRUARY 1, 1952–MAY 1, 1952. D. L. Foster and E. O. Nurmi. May 13, 1952. Decl. Feb. 13, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80 (mf OTS).

A procedure for dissolving thorium slugs was developed to meet specifications of the Interim 23 Process. The dissolvings were made to a 200% heel using approximately 5.2 moles of HNO_3 per mole of total metal to be dissolved. The feed acid to the dissolver was 13 N HNO_3 and 0.075 M in F^-. Dissolving time was 5 to 6 hours. It was found that with a dissolver equipped with an updraft condenser, approximately 0.5 moles of additional acid would be required per mole of thorium metal to be dissolved. This would indicate that with an updraft condenser, the original acid charge should be about 5.6 moles of acid per mole of total metal to be dissolved (aluminum plus thorium) instead of 5.2 as used with a downdraft condenser. About 10% of the weight of aluminum originally charged did not dissolve, but in subsequent runs, when additional canned slugs were added, the amount of undissolved aluminum did not change appreciably, but remained at about 10% of the original charge. (auth)

7485 CF-52-5-113
Oak Ridge National Lab., Tenn.
PLUTONIUM EXTRACTION FROM K-25 POT RESIDUE. Quarterly Report For Period February 10, 1952–May 10, 1952. J. R. Flanary. May 14, 1952. Decl. Feb. 13, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80 (mf OTS).

Plutonium analyses of the first sample of pot residue solution from K-25 showed that only 25% of the total Pu was extractable with TTA. Subsequent investigation under conditions designed to systematically eliminate the suspected sources of interference established that about 75% of the Pu in the sample had polymerized during several weeks storage in low acid. Although numerous attempts were made to break the polymer, none of the methods tried were sufficiently successful to permit quantitative extraction with TBP. It was found that Pu^{4+} hydrolysis could be avoided by adding 0.03M ferrous sulfamate to the ANN solution to reduce and hold Pu in the trivalent state until the pot residue solution could be withdrawn and adjusted to 1M HNO_3. (auth)

7486 CF-52-11-3
Oak Ridge National Lab., Tenn.
TESTING OF A SCALE MODEL $Ca(OH)_2$ CONTACTOR FOR THOREX SOLVENT RECOVERY. Quarterly Report for August 1, 1952 to November 1, 1952. P. A. Haas. Nov. 6, 1952. Decl. Feb. 16, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A scale model $Ca(OH)_2$ solvent purification contactor was set up and operated to study the design variables important for designing larger units for solvent recovery in the Thorex Process. Satisfactory mechanical operation of the contactor was verified, but the conditions for preparing and maintaining a $Ca(OH)_2$ slurry with the optimum properties were not fully determined. (auth)

7487 CF-53-3-246
Oak Ridge National Lab., Tenn.
CORROSION STUDIES FOR THE THOREX PROCESS. J. L. English. Mar. 27, 1953. Decl. Feb. 14, 1957. 21p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Stainless steel specimens were abraded and exposed to Thorex Process solutions and vapors at atmospheric and boiling temperatures. Weight losses and penetration measurements are recorded. (D.E.B.)

7488 CF-53-6-105
Oak Ridge National Lab., Tenn.
SOME INSTRUMENTATION DEVELOPMENT FOR

THOREX PULSE COLUMNS. Tom M. Gayle. June 12, 1953. Decl. Feb. 14, 1957. 13p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Special instruments were developed for and applied to the two-inch vertical pulse column and the two-inch concatenated pulse column of the Thorex process. The instruments were intended for a hot pilot plant operating under similar conditions. Instruments described include a pneumatic detector for interface level; a pneumatic detector for pulse amplitude (illustrated); an oscilloscope for pulse monitoring (illustrated); an instrument for interface level and amplitude detection by capacity measurement (illustrated); a flowmeter for radioactive liquids; and an automatic control for column feeds and withdrawals (illustrated). (C.H.)

7489 CF-53-6-110
Oak Ridge National Lab., Tenn.
A TEST OF THE $Ca(OH)_2$ SOLVENT RECOVERY SYSTEM ON PUREX FIRST URANIUM CYCLE SOLVENT. P. A. Haas. June 11, 1953. Decl. Feb. 14, 1957. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A vertical solid-liquid mixer-settler developed for semicontinuous recovery of used TBP solvents by contacting the solvent with a $Ca(OH)_2$ slurry effectively removed large amounts of crud forming materials present in used Purex first U cycle solvent from Purex Pilot Plant run HCP-20. This was evidenced by the $Ca(OH)_2$ shake test which indicated by the clarity of the solvent phase and lack of crud formation at the interface that the solvent had been stripped of all crud formers. This system for Purex solvent recovery is thus recommended for consideration as an alternative to the carbonatewash system for crud former removal. Decontamination of the Purex solvent was low. Irreversible combination of radioactive ions occurred with solvent degradation products which continued to build up during the two-month period following the Pilot Plant run and within which this test was made. Further chemical evaluation is required before a satisfactory indication can be obtained of the decontaminability of the $Ca(OH)_2$ procedure, and before a recommendation can be made concerning use of this system for decontamination. Ru and Zr were the chief activities left in the treated solvent. No I was detected in the solvent at the time of the test. (auth)

7490 CF-53-6-138
Oak Ridge National Lab., Tenn.
RECOMMENDED PULSE COLUMN DIMENSIONS AND ACCESSORIES FOR THE THOREX PILOT PLANT. A. C. Jealous. June 17, 1953. Decl. Feb. 20, 1957. 6p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

7491 CF-53-7-130
Oak Ridge National Lab., Tenn.
RUTHENIUM DISTILLATION IN PUREX ACID WASTE EVAPORATION. J. R. Flanary and V. J. Reilly. July 20, 1953. Decl. Feb. 14, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Ruthenium has been found to distill to a significant extent from boiling 7 N HNO_3. The formation of distillable Ru at this acidity requires an appreciable length of time, but is noticeable in 5 to 10 hours. There is some evidence from this work that Ru distills more readily from dilute acid containing $Fe_2(SO_4)_3$, H_2SO_4 and $NaNO_3$, the non-volatile components of Purex acid wastes. (auth)

7492 CF-53-9-188
Oak Ridge National Lab., Tenn.
RECOVERY OF 7500 AREA SALVAGE MATERIAL. S. Mann and J. E. Bigelow. Sept. 24, 1953. Decl. Feb. 13 1957. 18p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Salvage material from the 7500 Area, consisting mostly of ground-up concrete and other waste in nitric acid, was processed for recovery of its uranium content. The U^{235} content was originally estimated at 825.8 g, with an estimated isotopic purity of 92%. Of the total uranium actually present in pilot plant feed (1098.3 g) 97.9% was recovered, 1.16% was lost to samples and low activity waste streams, and 0.91% was lost to waste due to equipment failure. Processing rate was limited by the low flooding rate of the material, at times as low as 24 gal/hr/sq ft for some batches of feed. Column on stream time efficiency was 87%. The material was processed by solvent extraction using 5% tributyl phosphate (TBP) in Amsco diluent. Only slight modifications were required to adapt the equipment in Building 3503, formerly employed for the Interim 23 operation, to the new program. With the exception of minor mechanical interruptions and one major instance of column flooding, the plant operated smoothly five days per week from June 15 to July 30, 1953. The chemical flowsheet prescribed by the Chemical Development Section was based principally on a previous similar operation. In the present case, the uranium product was collected by ion exchange on two resin columns. Elution of the resin columns provided the final recovery figures: 1074.07 g of total uranium, including 945.66 g of U^{235} according to the average isotopic purity of 88.04% determined by mass spectrograph analysis. Analysis of adjusted pilot plant indicated 1322.2 g of total uranium; subtracting known operating losses and recycle material, the total uranium produced should be 1074.2 g, which is in good agreement with the actual amount obtained from the resin columns. (auth)

7493 CF-53-11-191
Oak Ridge National Lab., Tenn.
EXCER [PROCESS]: PRELIMINARY INVESTIGATION OF ELECTROLYTIC REDUCTION AND PRECIPITATION OF URANIUM FROM A NITRATE SYSTEM. E. D. Arnold and E. O. Nurmi. Nov. 20, 1953. Decl. Feb. 14, 1957. 17p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Several preliminary investigations indicated favorable results upon electrolyzing high (~50g/liter) uranium concentration solutions which yield black precipitates that remain black after washing. $NaNO_3$ and NH_4NO_3 systems were tried but NH_4NO_3 was found to be the most efficient. Cation membranes under favorable conditions produce the black precipitate but washing techniques will have to be improved to meet product specifications. (F.S.)

7494 CF-54-2-99
Oak Ridge National Lab., Tenn.
EXCER [PROCESS]: ANODE CORROSION IN AQUEOUS HYDROFLUORIC ACID ELECTROLYTES. J. H. Goode. Feb. 5, 1954. Decl. Feb. 14, 1957. 17p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

A major problem in the electrolytic reduction step in the Excer Process for the conversion of $UO_2(NO_3)_2$ to UF_4 is the stability of the anodes required in the reduction cell. A survey of possible anode materials was made in an effort to find the most suitable electrode for use in the electrolytic stage of the process. Results of the survey indicated that a number of grades of graphite were suitable for use as consumable anodes, under flowsheet conditions of 3M HF and boiling temperature, if the fine degradation particles can be kept from contaminating the U product slurry by a suitable wrapping or other means. Of the metallic anodes tested, commercially pure Pt appeared to have the smallest weight losses, but under the flowsheet conditions the losses were too high to make its use economically attractive. (auth)

7495 CF-54-2-130
Oak Ridge National Lab., Tenn.
MOVING BED–EXCER COST COMPARISON. F. E.

Harrington, J. E. Moore, and J. T. Roberts. Feb. 17, 1954. Decl. Feb. 14, 1957. 24p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

The results of a cost study made of the Moving Bed process and the Excer process on a comparable basis are given for a plant processing 22 tons of decontaminated U per day. The two processes are described and are illustrated by schematic diagrams. Detailed equipment costs are presented for both processes and are factored to obtain total capital investment. Operating costs, details, and summaries are included. (auth)

7496 CF-54-3-114
Oak Ridge National Lab., Tenn.
ANALYTICAL SCHEDULE FOR THOREX PILOT PLANT. C. E. Lamb, W. T. McDuffie, G. R. Wilson, and R. P. Wischow. Mar. 19, 1954. Decl. Feb. 14, 1957. 10p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

A review and revision of the schedule for analytical work related to the Thorex pilot plant were studied so that if the precision of the methods available will not meet requirements, new methods can be developed. (auth)

7497 CF-54-4-131
Oak Ridge National Lab., Tenn.
PREPARATION OF ACID DEFICIENT ALUMINUM NITRATE BY ELECTRODIALYSIS. E. J. Parsi. Apr. 14, 1954. Decl. Feb. 14, 1957. 14p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A continuous method for the production of acid deficient aluminum nitrate was developed in a semi-pilot scale electrodialysis unit. Using this method a clear solution of 90% monobasic 0.6M aluminum nitrate has been produced at a rate of 0.37 liters/hour. A by-product 0.29N nitric acid containing less than 40 ppm aluminum was also produced at a rate of 0.57 liters/hour. From the data obtained in this run, a preliminary cost estimate for the continuous production of acid deficient aluminum nitrate 0.6M in aluminum and 0.44 acid deficient was made. It was based on the requirements of the Thorex pilot plant of approximately 24 liters/hour. The cost is largely dependent on the rate of corrosion of the anode and on the labor required. However, on scaling up labor costs will be reduced, but the cost of the anode replacement will not be changed. The cost of a unit to satisfy the needs of the Thorex pilot plant is of the order of $500.00 exclusive of anode costs. Adding up the cost of auxiliary equipment and installation, the total cost of the unit is approximately $3700.00. It will cost $3500.00 to use 0.001 in. Pt foils as anodes. At this low scale of production and assuming a Pt corrosion rate of 0.05 mg/a-hour, the cost per pound of aluminum is approximately $0.64 of which $0.37 is labor and $0.10 is Pt loss. To this must be added $1.49/lb of Al in the aluminum nitrate feed. Scaling up by a factor of 20 over the Thorex pilot plant unit, the installed cost of the unit will be of the order of $43,500, and the investment in platinum about $70,000. Using the same assumption as in the smaller scale unit, the cost per pound of Al produced is approximately $0.35, of which $0.11 is labor and $0.10 is Pt loss. Again the cost of Al in the feed must be added to the above figure. All the equipment was amortized at 16% per annum, except for the ion exchange membranes and gaskets, for which a conservative 200% was used. This method is very attractive, but its economic feasibility hinges heavily on the rate of corrosion of Pt as an anode in nitric acid or in the development of a suitable substitute anode. (auth)

7498 CF-54-9-171
Oak Ridge National Lab., Tenn.
URANIUM ISOTOPE FRACTIONATION FACTOR IN BATCH EQUILIBRATION BETWEEN RESIN AND VERSENE SOLUTION. J. S. Drury and A. C. Rutenberg. Sept. 29, 1954. Decl. Feb. 14, 1957. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The isotopic fractionation factor for the reaction of aqueous uranyl ion (in the presence of diammonium versenate ion) with the uranyl form of Dowex-50 resin was estimated to be 1.00006 for the following reaction: $U^{238}O_2^{2+}$ (aqueous versenate) + $U^{235}O_2R_2 \rightleftharpoons U^{235}O_2^{2+}$ (aqueous versenate) + $U^{238}O_2R_2$, where R is the resin. (C.W.H.)

7499 CF-54-11-117
Oak Ridge National Lab., Tenn.
PRELIMINARY STUDY OF "COTTON" AND THORIUM "BACKGROUNDS" IN THE THOREX PROCESS. S. A. Reynolds. Nov. 22, 1954. Changed from SECRET Dec. 16, 1955. Decl. Feb. 14, 1957. 14p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The natural beta and gamma backgrounds associated with Th and U^{233} in the Thorex Process have been approximated. Corrections were made for the self-absorption of U^{233} betas in Th and in $SnCl_2$. (C.W.H.)

7500 CF-55-1-45
Oak Ridge National Lab., Tenn.
UNIT OPERATIONS STATUS REPORT FOR DECEMBER 1954. PART I. W. K. Eister. Jan. 6, 1955. Decl. Feb. 15, 1957. Contract [W-7405-eng-26]. $0.35(OTS).

Homogeneous Reactor. Loop studies were continued on rare earth sulfate stability under reactor operating conditions, and on the isolation of solids to make physical measurements of their size, density, and adhesive properties. Tests were carried out to determine optimum size and configuration of the hydroclones. The recovery of D_2O by evaporation was discussed and operating conditions for the evaporator outlined. (C.W.H.)

7501 CF-55-1-153
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR PROCESSING. HRT AND TBR BATCH HYDROCLONE RECOMMENDATION. C. E. Guthrie. Jan. 20, 1955. Decl. Feb. 20, 1957. 17p. Contract W-7405-eng-26. $0.30(OTS).

A single-stage hydroclone and batch letdown system have been found desirable for the TBR. The projected requirements for the TBR would dictate a system of a single stage hydroclone, at 3 gpm feed rate, with a 5.3 gallon retention tank dumped every four hours. Therefore, a similar system should be installed in the HRT to demonstrate the equipment. A two gallon retention tank and a single hydroclone at 0.2 gpm feed rate are recommended for initial installation with replacement by a 1 gpm hydroclone stage when the HRT is expanded to 30 megawatts. A summary of the operating conditions will be found in Table I. The hydroclone used will depend on future hydroclone development and precipitate particle size. It may consist of a single hydroclone or multiple smaller units in parallel. (auth)

7502 CF-55-3-120
Oak Ridge National Lab., Tenn.
RECOVERY OF U-235 FROM HRE FUEL BY THE INTERIM-23 FACILITY. K. H. McCorkle. Mar. 17, 1955. Decl. Feb. 14, 1957. 20p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The recovery and decontamination of U^{235} from spent HRE fuel solution by solvent extraction with TBP was investigated. Recoveries of 91.1% U^{235} were obtained. The gross β and gross γ-decontamination factors were 2.8×10^5 and 4.9×10^4 respectively. (C.W.H.)

7503 CF-55-4-149
Oak Ridge National Lab., Tenn.
25 PROCESS ASSISTANCE. For Period of April 4, 1955 To April 15, 1955. F. L. Culler. Apr. 26, 1955. Decl. Feb.

14, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Work on organic emulsion formation under 1A column feed-plate conditions was continued. Results of tracer level runs indicate that Zr and Nb were the principal contaminants in the first cycle U product. Thorium oxychloride preparation, thorium nitrate chlorination, thorium(IV) chloride reduction with Mg, and Th dissolution in Hg were investigated. (C.W.H.)

7504 CF-55-5-137
Oak Ridge National Lab., Tenn.
25 PROCESS ASSISTANCE. For Period of May 2, 1955 to May 13, 1955. F. L. Culler. May 23 1955. Decl. Feb. 14, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Studies were continued on the radiation damage in recycled TBP solvent, organic emulsion formation in Al solutions, chlorination of thermally denitrated thorium nitrate hydrate, and reduction of $ThCl_4$ with sodium amalgam. (C.W.H.)

7505 CF-55-7-49
Oak Ridge National Lab., Tenn.
THOREX SECOND THORIUM CYCLE. R. P. Wischow. July 5, 1955. Decl. Feb. 14, 1957. 11p. Contract [W-7405-eng-26]. $0.25(OTS).

Operating data and flowsheets are given for the Thorex second thorium cycle. This cycle is being added to the original Thorex extraction system to ensure adequate Th decontamination. (C.W.H.)

7506 CF-56-1-151
Oak Ridge National Lab., Tenn.
THE HERMEX PROCESS FOR METAL DECONTAMINATION BY MERCURY PROCESSING. B. H. Morrison and R. E. Blanco. Jan. 25, 1956. Decl. Feb. 14, 1957. 19p. Contract [W-7405-eng-26]. $0.30(OTS).

The Hermex Process for the decontamination of irradiated metals and purifying scrap virgin metals is presented. The process as applied to irradiated U consists of the following steps: the irradiated U is continuously dissolved in boiling Hg at 365°C; the hot U–Hg solution from the dissolver vessel is withdrawn continuously leaving a slag layer containing ~87% of the fission products; the U–Hg solution is cooled to 25°C and concentrated by vacuum filtration; the U quasi-amalgam is washed with dilute HCl to remove the fission products; the Hg is volatilized from the quasi-amalgam; and the U melted to dense metal by slow heating to 1250°C in a single operation. (W.L.H.)

7507 CF-56-1-175
Oak Ridge National Lab., Tenn.
CHEMICAL TECHNOLOGY DIVISION, UNIT OPERATIONS SECTION MONTHLY PROGRESS REPORT [FOR] JANUARY 1956. W. K. Eister. Decl. Feb. 14, 1957. 49p. Contract [W-7405-eng-26]. $0.45(OTS).

7508 CF-56-3-131
Oak Ridge National Lab., Tenn.
PROGRAM FOR THE HRT CORE PROCESSING PLANT. R. B. Lindauer. Mar. 21, 1956. Decl. Feb. 13, 1957. 8p. Contract [W-7405-eng-26]. $0.25(OTS).

An outline of the program for the HRT core processing plant is presented. The program consists of cold shakedown operation–pre-operational checks, water operation (high and low pressure systems), remote maintenance tests, system cleanup, operation with simulated soup, and preparation for hot operation and hot operation–saturation of chemical plant equipment and removal of solids by hydroclone and hot surfaces. A brief description of the operating procedure is included. (C.W.H.)

7509 CF-56-5-48
Oak Ridge National Lab., Tenn.
POWER REACTOR FUEL PROCESSING PRELIMINARY REPORT. R. E. Blanco, W. K. Eister, and D. E. Ferguson. May 14, 1956. Decl. Feb. 20, 1957. 33p. Contract [W-7405-eng-26]. $0.35(OTS).

Fused Salt–Fluoride Volatility Process. An over-all decontamination factor of more than 10^5 and a U loss of less than 0.1% have been obtained with the fluoride volatility process in laboratory tests. General Dissolution Studies. Fuels containing stainless steel were carburized or dissolved in molten Al and then dissolved in HNO_3. Such fuels were dissolved directly in aqua regia, H_2CrO_4 – HNO_3, or, by anodic electrolysis, in HNO_3. Uranium was recovered from Zr fuels by leaching with HNO_3 after oxidation, Zn or Al alloying, fused salt dissolution, or electrolytic dissolution. Molybdenum–uranium alloy was directly soluble in HNO_3 at acceptable rates. The rates of dissolution of U alloys with Zr, Nb, Si, and Mo in HNO_3 or dilute aqua regia are listed. Solvent Extraction Studies. Adaptations of solvent extraction processes now in use appear feasible for processing solutions of fuel elements containing difficultly soluble metals. Zircex Process. Hydrochlorination of a Zr-containing fuel element produced $ZrCl_4$, which volatilized. Uranium chloride in the residue readily dissolved in HNO_3. Hermex Process. Massive U dissolved rapidly in boiling Hg and was recovered from the cooled amalgam as uranium mercuride by filtering and pressing. Fission product decontamination factors were 10^2 to 10^3 for 3-year-decayed U and 15 to 28 for 45-day-decayed U. The uranium mercuride was converted to U metal by volatilizing the mercury and melting the U in a single step at 1200°C. (auth)

7510 CF-56-6-27
Oak Ridge National Lab., Tenn.
HRP-BP: DISSOLVING SOLUTION FOR SOLIDS REMOVED FROM UO_2SO_4 BLANKET OF HRT. William L. Carter. June 4, 1956. Decl. Feb. 14, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

When the HRT is converted from a heavy water blanket to a UO_2SO_4 blanket, a processing plant will begin operation to remove the Pu, fission, and corrosion products from the blanket. A short discussion of the expected corrosion rates of the stainless steel parts in the blanket system and the composition of the blanket before processing is presented. (W.L.H.)

7511 CF-56-6-100
Oak Ridge National Lab., Tenn.
FEED MATERIALS PROCESSING STATUS REPORT FOR JUNE 1956. R. E. Blanco and W. K. Eister. June 15, 1956. Decl. Feb. 14, 1957. 22p. Contract [W-7405-eng-26]. $0.30(OTS).

Metallex Process. Methanol and petroleum ether from which dissolved O_2 has been removed have less oxidative effect upon Th amalgam than water with or without dissolved O_2. A rough correlation between the free voidage of sintered Th and its oxide content, and therefore the metal quality, is indicated. A procedure was developed for determining the total voidage and the atmosphere-connected voidage in sintered Th. Fluorox Process. The specific surface area of UF_4 appears to be constant over the temperature range 25 to 600°C. Above 600°C the specific surface area decreases as sintering begins. The major impurities found in UF_6 produced from African ore concentrate were Al, Ca, Cu, and Fe. The reduction-hydrofluorination and oxidation reactors were modified to improve their operability, and 2 runs were made with the improved oxidation reactor. Low-carbon UF_4, suitable for fluorination to UF_6 or reduction to metal, was prepared by

burning the carbon in the UF_4 with O_2. Excer Process. Results of runs in the 2-in. diameter Higgins continuous ion-exchange contactor indicate that UO_2Cl_2 product may be withdrawn at concentrations as high as 140 g of U per liter and containing as low as 0.05M NaCl. Conductivity instruments provided excellent process control, which in turn allowed efficient usage of 5M NaCl eluant. This eliminated the need for recycle of the NaCl. The presence of Na or Ca did not affect the density of the electrolytic $UF_4 \cdot {}^3/_4H_2O$ product. An electrolytic cell with 64 $in.^2$ inside area was operated 5 hr to produce $UF_4 \cdot {}^3/_4H_2O$. The resulting slurry of $UF_4 \cdot {}^3/_4H_2O$ appears acceptable as far as physical characteristics are concerned for making specification grade UF_4 after filtration and dehydration. A nickel ribbon flite dehydrator is being constructed. (For preceding period see CF-56-5-100.) (auth)

7512 CF-56-7-100
Oak Ridge National Lab., Tenn.
FEED MATERIALS PROCESSING STATUS REPORT FOR JULY 1956. R. E. Blanco and W. K. Eister. July 15, 1956. Decl. Feb. 14, 1957. 19p. Contract [W-7405-eng-26]. $0.30(OTS).

Metallex Process. The effect of a small addition of Mg to Th quasi-amalgam before and after reduction is reported along with the relation between free voidage of the product Th and its quality. Fluorox Process. The results of studies on C removal from UF_4 with H_2 and on surface areas of UO_2F_2 and UF_4 and on the isolation of an unknown compound are given. New equipment and modifications are reported. Excer Process. The application of the process to ore processing is being investigated and the results of some experiments and tests are reported. (T.R.H.)

7513 CF-56-7-101
Oak Ridge National Lab., Tenn.
POWER REACTOR FUEL PROCESSING STATUS REPORT FOR JULY 1956. R. E. Blanco, W. K. Eister, and D. E. Ferguson. July 15, 1956. Decl. Feb. 14, 1957. 23p. Contract [W-7405-eng-26]. $0.30(OTS).

Fused Salt — Fluoride Volatility Process. Information on the corrosion of Ni, rate of dissolution of Zr–U alloy in fused $NaF-ZrF_4$, fluoridation of molten $NaF-ZrF_4-UF_4$, and decomposition of $UF_6 \cdot 3NaF$ is reported. Zircex Process. Data on hydrochlorination rates, rare earth separations, fuel element core leaching, heat transfer problems, and sublimed $ZrCl_4$ are included. Hermex Process. Dissolution of U in Mg amalgam and U in Bi amalgam are reported. General Dissolution Studies. Dissolution studies of Zircaloy-2 in aqueous $FeCl_3$ and $FeCl_3$–NaCl and HCl were made. Cladding removal is included. Mechanical Dejacketing and Handling Studies. A mechanical decladder and a leaching-dissolving device are being investigated. Materials Evaluation. Results of tests of materials for resistance to HNO_3 are reported. (T.R.H.)

7514 CF-56-8-100
Oak Ridge National Lab., Tenn.
FEED MATERIALS PROCESSING STATUS REPORT FOR AUGUST 1956. R. E. Blanco and W. K. Eister. Aug. 15, 1956. Decl. Feb. 20, 1957. 18p. Contract [W-7405-eng-26]. $0.30(OTS).

Metallex Process. Preliminary experiments indicated that aqueous washing of the Th amalgam may be eliminated by dissolution of the $ThHg_x$ in the Hg phase and filtering while hot to separate it from undissolved impurities. Further work with antioxidants in an aqueous wash solution showed that they consistently lower the quality of the metal. X-ray-diffraction studies of Th amalgam showed that the species of $ThHg_x$ produced are dependent on time and temperature, but not on washing technics. Installation of the interim facility for larger-scale production of Th quasi amalgam by the dry salt flowsheet was completed. Fluorox Process. The rate of oxidation of UF_4 by air increased rapidly as the temperature was increased from 700 to 800°C. In preliminary oxidation runs in the lined moving-bed reactor, in which graphite pellets were used as the source of the heat, the UF_4 pellets fused. An unlined fluidized bed was designed in which temperature control should be more precise. Excer Process. The high decontamination factors obtained in electrolysis of a solution of crude ore concentrate produced by solvent extraction indicated that in some cases the ion-exchange step may be omitted. (auth)

7515 CL-P-368
Clinton Labs., Oak Ridge, Tenn.
SEMIMONTHLY REPORT AND MANPOWER DISTRIBUTION FOR PERIOD DECEMBER 1 TO 15, 1944. S. G. English. Dec. 16, 1944. Decl. Feb. 16, 1957. 43p. Contract [W-7405-eng-39]. $7.80(ph OTS); $3.30(mf OTS).

Studies on Pu carrying in the $BiPO_4$ process include the effect of N_2H_4 and methods of destroying N_2H_4 on Pu carrying, effect of storage conditions on the preextraction treatment needed, and analytical methods for N_2H_4. The oxalate crossover was investigated in studies of decontamination in the process. The reaction of Pu in alkaline solutions, solubility of $Pu(SO_4)_2$, and effect of various elements on peroxide precipitations in the Hanford isolation step were investigated. Miscellaneous studies included Pu loss in reprecipitations, precipitating reagents, removal of phosphate, corrosion of 25-12 stainless steel in HNO_3–HF and $HNO_3-H_2SiF_6$ Th dissolver solutions and U determination in presence of $Th(NO_3)_4$. (G.Y.)

7516 CN-1071
[Chicago. Univ.] Metallurgical Lab.
CHEMICAL RESEARCH — PRODUCTION AND EXTRACTION OF PLUTONIUM; REPORT FOR PERIOD ENDING NOVEMBER 11, 1943. Decl. Feb. 14, 1957. 27p. Contract [W-7401-eng-37]. (A-1459). $4.80(ph OTS); $2.70 (mf OTS).

Progress is reported in studies on adsorption methods for the extraction and decontamination of Pu. Data are included from studies on adsorption on Ti zeolite, calcined zeolite, and Amberlite IR-1. (C.H.)

7517 CN-1227
Chicago. Univ. Metallurgical Lab.
PROGRESS REPORT ON THE ADSORPTION EXTRACTION PROCESS FOR THE TWO MONTH PERIOD ENDING DECEMBER 1, 1943. A. R. Brosi and E. R. Russell. Decl. Feb. 16, 1957. 19p. (A-1806). $3.30(ph OTS); $2.40(mf OTS).

7518 CN-1409
Clinton Labs., Oak Ridge, Tenn.
SEPARATIONS DEVELOPMENT DIVISION SEMIMONTHLY REPORT FOR PERIOD ENDING MARCH 31, 1944. O. H. Greager. Decl. Feb. 18, 1957. 11p. $3.30(ph OTS); $2.40(mf OTS).

Studies were continued on the metal solution, extraction, crossover, and decontamination steps of the Bismuth Phosphate Process. (C.W.H.)

7519 CN-1412
Clinton Labs., Oak Ridge, Tenn.
OXALATE AS A REDUCING AGENT. Final Report [on] Problem Assignment No. 203-X27C. Apr. 22, 1944. Decl. Feb. 16, 1957. 9p. (A-2277). $1.80(ph OTS); $1.80(mf OTS).

In the reduction of Pu by $H_2C_2O_4$ in the crossover cycle and the LaF_3 concentration cycles, Mn ions catalyzed the reduction and rapid rates were obtained at 35°C. Reductions were carried out in the crossover cycle (no F by-

product) with oxalic acid alone and uranous oxalate was precipitated directly as an alternate cross-over carrier. (F.S.)

7520 CN-1641
Clinton Labs., Oak Ridge, Tenn.
SURFACE REACTIONS OF ZIRCONIUM, BARIUM AND TELLURIUM IONS WITH LANTHANUM FLUORIDE AND MANGANESE DIOXIDE. Summarizing Report: Part I. Problem Assignments: 205-X48C and 203-X72C. Howard Gest and W. H. Burgus. July 7, 1944. Decl. Feb. 16, 1957. 23p. Contract [W-7401-eng-37]. $4.80(ph OTS); $2.70 (mf OTS).

Surface adsorption was found to be largely responsible for the coseparation of tracer 65 d zirconium activity in 1M HNO_3 + 0.5M HF on both (a) externally and (b) coformed (La^{3+} last) LaF_3. The Langmuir adsorption equation, $(x/m) = k_1c/(1 + k_2c)$, served to describe the data where in (a) $k_1 = 0.05$, $k_2 = 2.3$ and in (b) $k_1 = 0.05$, $k_2 = 0.7$, respectively. Further, the adsorption of Zr was found to be reversible, to be independent of the concentration of F^- from 0.1 to 1.0M and of HNO_3 from 0.3 to 1.0M, and, to be unaffected by temperature from 25 to to 70°C. It was possible to calculate a specific surface of 58 m^2/gm for the LaF_3 from the observed saturation adsorption of 14.6×10^{19} fluozirconate ions per gram. This value compared favorably with that of 59 m^2/gm determined by the van der Waals adsorption of N_2 at –195.8°C. These results indicated the feasibility of the addition of "hold-back" Zr at 1 to 2 mg/ml. to obtain a 30-fold improvement in the specific decontamination of Zr in the first product carrying LaF_3. This level of zirconium gave waste losses of the order of 2% in 330S solutions. The coseparation of tracer 12.5 d barium activity with externally formed LaF_3, and the carrying down of zirconium and tellurium (32 d) activities with externally prepared MnO_2 could not be explained by any simple mechanism, although the empirical Freundlich equation, $(x/m) = kc^n$ appeared to fit the data for the first two systems, where k = 2.8, 0.7 and n = 0.67, 0.43, respectively. (auth)

7521 CN-1839
Clinton Labs., Oak Ridge, Tenn.
FISSION PRODUCT DISTRIBUTION IN AN ADSORPTION EXTRACTION DECONTAMINATION CYCLE WITH AN ION EXCHANGER. J. A. Swartout and D. N. Hume. July 10, 1944. Decl. Feb. 14, 1957. 15p. Contract [W-7401-eng-37]. (A-2496). $3.30(ph OTS); $2.40(mf OTS).

The adsorption extraction process in which Pu and U are adsorbed from 10% UNH solutions by an ion-exchanger; adsorbed U is removed by 0.25M H_2SO_4; adsorbed Pu is desorbed with 1.25M $NaHSO_4$; and the residual strongly adsorbed fission product activity is removed with 0.4% $H_2C_2O_4$, was modified to include a separate by-product desorption step. This change resulted in the improvement of the gross β and γ decontamination factors from 2 to 75, and from 4 to 95, respectively. (auth)

7522 CN-2511
[Chicago. Univ. Metallurgical Lab.]
SOLVENT EXTRACTION PROCESS FOR CONCENTRATION AND ISOLATION OF PRODUCT. Progress Report. Problem Assignments 209-MLC-1401, 343-MLC-1402, and 209-MLC-1706. S. Lawroski, L. R. Dawson, and J. B. Tepe. Dec. 30, 1944. Decl. Feb. 16, 1957. 40p. Contract W-7401-eng-37. $6.30(ph OTS); $3.00(mf OTS).

The progress of the work from July 1944 to January 1945 on the development of the hexone solvent extraction process for concentrating and isolating Pu is reported. (F.S.)

7523 HW-14923
Hanford Works, Richland, Wash.
CORROSION RATES OF MILD AND STAINLESS STEELS EXPOSED IN REDOX STREAM IAW (ANL, JUNE 1, 1948 FLOWSHEET). W. W. Koenig. Nov. 1, 1949. Decl. Feb. 20, 1957. 12p. Contract [W-31-109-eng-52]. $3.30(ph OTS); $2.40(mf OTS).

A test program has been carried out to determine the relative corrosion resistance of welded mild and stainless steels in Redox Stream IAW neutralized to pH 0, 2, and 10. Stainless steel types T-304 and T-347 show excellent resistance to Redox Stream IAW neutralized to pH 0, 2, and 10. Mild steel type SAE 1020 shows good resistance to this stream at pH 10 and very poor resistance at pH 2 and 0. (auth)

7524 HW-17226
Hanford Works, Richland, Wash.
SOLUBILITY RELATIONSHIPS OF ACIDIFIED STORED METAL WASTES. M. H. Curtis. Mar. 24, 1950. Decl. Feb. 20, 1957. 16p. Contract [W-31-109-Eng-52]. $3.30 (ph OTS); $2.40(mf OTS).

The preparation of feed (RAF) solution for the TBP solvent-extraction process from aged neutralized metal waste relies upon acid dissolution of the solid phases (Na uranyl carbonates and Na uranyl phosphates) which are present in the storage tanks. Through experimental investigation it has been found possible to define the solubility relationships of the two solid phases (uranyl hydrogen phosphate and $NaNO_3$) which occur in the acidified system as a function of the acidity in a system containing the components (uranyl, Na^+, H^+, phosphate, sulfate, and nitrate ions) of the feed solution formed in the manner indicated. The range of the acid concentrations studied is believed to embrace all those of interest in the preparation of the feed stream (RAF) and those of the aqueous phase at any point within the extractant (RA) column or its effluent aqueous (RAW) stream. (D.E.B.)

7525 HW-17698
Hanford Works, Richland, Wash.
FERROUS SULFAMATE STABILITY IN THE RA SYSTEM. F. Clagett. Apr. 20, 1950. Decl. Feb. 19, 1957. 10p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

In the absence of a stabilizing agent Fe^{2+} is oxidized immediately in 5M HNO_3. Sulfamic acid or sulfamate ions added to the solution increase the Fe^{2+} stability (~30 hr). Sulfamate ion stability tests were inconclusive. (C.W.H.)

7526 HW-17981
Hanford Works, Richland, Wash.
CORROSION OF AUSTENITIC STAINLESS STEEL IN MIXED ACIDS. W. W. Koenig. June 5, 1950. Decl. Feb. 19, 1957. 8p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

An investigation was made of the corrosion rates of several austenitic stainless steels in simulated Redox Process waste storage solution containing HF, HNO_3, and H_2SO_4 acids. Carpenter 20 and T-309 SCb steels showed good resistance to solutions of two concentrations, while T-304 ELC and T-347 showed poor resistance to one concentration and good to fair resistance to the other. (B.J.H.)

7527 HW-18149
[Hanford Works, Richland, Wash.]
SLUDGE ACIDIFICATION. M. K. Harmon. June 23, 1950. Decl. Feb. 19, 1957. 12p. Contract [W-31-109-eng-52]. $3.30(ph OTS); $2.40(mf OTS).

Problems of sludge-supernate acidification applicable to the TBP Process were surveyed and available information

was summarized. Data are tabulated from a study on the acidification of a synthetic incubated metal waste, free of product and fission products. The physical properties of the slurry before and after acidification, and the effects of heat evolved during acidification, and rate of slurry addition were studied. (C.H.)

7528 HW-19105
Hanford Works, Richland, Wash.
CHLORIDE REMOVAL FROM METAL WASTE SOLUTIONS. L. H. Clark. Oct. 10, 1950. Decl. Feb. 19, 1957 13p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40 (mf OTS).

Chloride ion in the RAW waste stream from the TBP metal recovery process would excessively corrode the stainless steel in a HNO_3 recovery step. The conditions under which peroxysulfate ($K_2S_2O_8$) will oxidize quantitatively the Cl^- to Cl_2, which may be removed by air sparging, are presented in detail. A preliminary survey of other oxidants was made. The results of this survey are tabulated, as are the per cent decomposition of $S_2O_8^{2-}$ by synthetic RAW, formation of ClO_3 and ClO_4 during Cl^- removal. (C.H.)

7529 HW-27489
[Hanford Atomic Products Operation, Richland, Wash.]
FACTORS AFFECTING THE CONCENTRATION OF TBP PLANT WASTES. D. F. Shepard. Mar. 25, 1953. Decl. Feb. 19, 1957. 11p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

Control of neutralized waste pH at 9.5 has relieved the process difficulty of line plugging in the TBP plant waste disposal. (F.S.)

7530 IDO-14312
Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
PULSE COLUMN STUDIES USING STR AND SIR FEED SOLUTIONS. S. L. Friederichs, D. E. Griffin, G. E. Lohse, and C. E. Stoops. Sept. 27, 1954. Decl. Feb. 28, 1957. 42p. Contract AT(10-1)-205. $6.30(ph OTS); $3 00 (mf OTS).

Pilot plant studies were made to determine operating characteristics of pulsed columns proposed as the solvent extraction contactors for the recovery of uranium from spent STR and SIR feed assemblies. Twenty-five STR "A" Column runs, six SIR "A" Column runs, one "B" Column run, and eight "C" Column runs were made. Variation of pulse conditions resulted in HTU values for the STR "A" Column ranging from 2.2 to 11.6 ft. Reduction of the aqueous-to-organic flow ratio from the original flowsheet value of 4.5/1 to 1.4/1 resulted in HTU values as low as 1.5 ft. The HTU values for the SIR "A" Column ranged from 1.0 to 1.6 ft. The HTU values for the "C" Column ranged from 1.5 to 1.9 ft. The maximum flooding volume velocity was found to be 1300 gal/(hr)(sq.ft.) for the "A" Column on both the STR and SIR flowsheets. The maximum "B" Column flooding point was found to be 800 gal/(hr) (sq.ft.) The "C" Column flooding points were found to be dependent upon nitric acid concentration in the organic phase. The maximum flooding point varied from 800 gal/ (hr)(sq.ft.) for 0.002 molar acid feed to 1100 gal/(hr)(sq.ft.) for 1.0 molar acid feed. (auth)

7531 IDO-14358
Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
EFFECT OF FIRST CYCLE SCRUB RATIO REDUCTION ON DECONTAMINATION IN THE "25" PROCESS. F. P. Schilling. Dec. 1, 1955. Decl. Feb. 27, 1957. 15p. Contract AT(10-1)-205. $0.25(OTS).

The scrub flow ratio in the first cycle of the hexone solvent extraction process for the recovery and decontamination of U from Al–U alloy fuels has been reduced by a factor of 10. A minimum flow ratio of scrub:feed: extractant of 1:79:67 has been plant tested. Uranium losses were unaffected, and overall fission product decontamination for three cycles was changed only slightly. (auth)

7532 ISC-234
Ames Lab., Ames, Iowa.
RECOVERY OF TRIBUTYL PHOSPHATE SOLVENT EMPLOYED IN THE EXTRACTION SEPARATION OF HAFNIUM AND ZIRCONIUM. W. R. Millard and R. P. Cox. [1952?]. Decl. Feb. 26, 1957. 32p. Contract W-7405-eng-82. $0.35(OTS .

A process for the production of Hf-free Zr by extraction with a TBP solvent containing dibutyl ether as a diluent is under development. The process involves the separation of hafnyl and zirconyl nitrates and chlorides in a mixer-settler extractor, producing an organic solvent containing purified Zr. The purified Zr is removed from the tributyl phosphate-dibutyl ether solvent by water stripping in a mixer-settler extractor. Approximately 5 stages are required to completely strip the solvent of the 0.5M zirconium originally present and to reduce the nitric acid concentration from 2.50M to about 0.1M with a water-to-solvent volume ratio of 0.8. The degradation of the solvent that appeared with extended use was eliminated by two alternative regeneration methods. A batch method using H_2SO_4 and Na_2CO_3 washes has been thoroughly tested and found to be effective. A continuous regeneration method has been developed which has many advantages over the batch procedure. (auth)

7533 KAPL-89
Knolls Atomic Power Lab., Schenectady, N. Y.
SULFAMIC ACID IN THE REDOX PROCESS. E. Zebroski and R. C. Feber. July 15, 1948. Decl. Feb. 26, 1957. 21p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

Experimental data and literature pertinent to the use of sulfamic acid in the Redox Process are summarized. (B.J.H.)

7534 KAPL-314
Knolls Atomic Power Lab., Schenectady, N. Y.
A SURVEY OF DEVELOPMENTS IN PLUTONIUM PROCESS CHEMISTRY FOR 1948-1949. H. Ward Alter and Edwin L. Zebroski. Mar. 17, 1950. Decl. Feb. 26, 1957. 63p. Contract W-31-109-eng-52. $0.50(OTS).

The work done at various times during 1948 and 1949 at Knolls Atomic Power Lab. and a condensed survey of work at Argonne, Hanford, and Oak Ridge on the chemistry of Pu in solvent extraction are presented, particularly in connection with the Redox Process. (L.M.T.)

7535 KLX-59
Kellex Corp., New York.
INVESTIGATION OF CENTRIFUGAL EXTRACTORS. Detailed Project Status Report on Job 11. Sept. 16, 1949. Decl. Mar. 7, 1957. 43p. Contract AT-30-1-GEN-169. $9.30(ph OTS); $3.60(mf OTS).

The main effort of the project was directed toward a complete investigation of centrifugal extractors, and included a literature and patent search, theoretical studies, studies of existing devices, a test program, and development and construction of actual models. This work was carried on as a search for a compact, easily operated centrifugal extractor which could be used as the extraction unit in a large-scale chemical processing plant. On the basis of the work done on this project, it was concluded that a non-rotating type of centrifugal extractor, which was designed, constructed and tested, is the most promising centrifugal type presently available. (auth)

7536 KLX-1741
Vitro Labs., West Orange, N. J.
HOMOGENEOUS REACTOR FUEL REPROCESSING. Quarterly Progress Report [for] October 1–December 31, 1954. Job 1087. Jan. 24, 1955. Decl. Feb. 28, 1957. 13p. [For Oak Ridge National Lab.]. Contract W-7405-eng-26, Subcontract No. 535. $0.25(OTS).

A high temperature loop was designed and operated for studying the precipitation of insoluble reactor products in a dynamic system. The equilibrium relationships concerning a process for removing I_2 from the core system by the reaction of metallic Ag with molecular I_2 in the vapor phase were studied. An inorganic cation exchanger (Zr phosphate) was investigated as an adsorber of Cs. (J.E.D.)

7537 MITG-520
Massachusetts Inst. of Tech., Watertown, Mass.
Mineral Engineering Lab.
PROGRESS REPORT ON DOMESTIC ORES FOR FEBRUARY 16, 1950 TO MARCH 15, 1950. John Dasher. Mar. 15, 1950. Decl. Mar. 8, 1957. 9p. Contract AT-30-1-Gen-211. $1.80(ph OTS); $1.80(mf OTS).

About 90% of U is extracted from Anaconda acid by one-step treatment with recycled monazite. An acetone wash after an ether leach recovers the rest of the U and nearly all other nitrates from the residue. Fractionation and metallurgical tests run on Marysvale and Sunshine ores are briefly discussed. (auth)

7538 MITG-525
Massachusetts Inst. of Tech., Watertown, Mass.
Mineral Engineering Lab.
PROGRESS REPORT ON DOMESTIC ORES FOR JULY 16, 1950 TO AUGUST 15, 1950. John Dasher. Aug. 15, 1950. Decl. Mar. 7, 1957. 13p. Contract AT-30-1-Gen-211. $3.30(ph OTS); $2.40(mf OTS).

Excellent results were obtained by ion exchange of Marysvale and Sunshine solutions. Loading on the resin is good with long retention times. Good U extraction was obtained in acid leaching the first sample of Cu – U ore. When $Ca(NO_3)_2$ is used in solvent leaching of damp sulfated carnotites, U extractions up to 98% are obtained. (auth)

7539 MonN-125
Clinton Labs., Oak Ridge, Tenn.
SEPARATION OF U^{233} FROM THORIUM. Problem Assignment No. TX5-11. Progress Report for the period July 1, 1945 to June 1, 1946. F. L. Steahly. July 26, 1946. Decl. Feb. 28, 1957. 62p. Contract W-7405-eng-39. $10.80(ph OTS); $3.90(mf OTS).

Flowsheets for two alternative processes for U^{233} and Pa^{233} recovery and decontamination based on batch extraction and continuous extraction are presented. The batch process flowsheet includes an extraction step with diisopropyl ether and a by-product $Th(C_2O_4)_2$ precipitation for U^{233} separation. The process for Pa^{233} recovery consists of two solvent extractions with diisopropyl carbinol on the aqueous waste from the first U^{233} extraction. (C.W.H.)

7540 MonN-127
[Clinton Labs., Oak Ridge, Tenn.]
ISOLATION OF U^{233} FROM THORIUM AND ITS COMPOUNDS. SEPARATION AND PURIFICATION OF U^{233} FROM SOLUTION, CONTAINING THORIUM NITRATE BY CONTINUOUS COUNTERCURRENT EXTRACTION IN A PACKED COLUMN. Progress Report—P. A. No. CX9-1. C. F. Booth. Sept. 16, 1946. Decl. Mar. 4, 1957. 57p. Contract W-35-058-eng-71. $10.80(ph OTS); $3.90(mf OTS).

The continuous countercurrent extraction of uranium by dibutyl cellosolve from solutions containing thorium nitrate, together with nitrates of aluminum, ammonia, calcium, and hydrogen, was investigated on a laboratory scale. Thorium was removed from the dibutyl cellosolve extract by continuous countercurrent scrubbing with calcium nitrate solution in an upper section of the column. The efficiency of several types of column packing was investigated. The packings which were studied were $\frac{1}{16}$ in. glass helices, $\frac{1}{8}$ in. glass helices, and split 6 mm glass rings. Using different packings, the HETS was determined for both the extraction step and the scrubbing step at several rates of flow with the aqueous phase continuous and also with the solvent phase continuous. Higher rates of flow invariably resulted in larger HETS values. In the case of the $\frac{1}{16}$ in. glass helices, the HETS was lower with the solvent phase continuous. With the split 6 mm glass rings it was lower with the aqueous phase continuous. With $\frac{1}{8}$ in. glass helices, there was little difference in HETS from the choice of continuous phase. With the same packing and at similar rates of flow, the HETS for scrubbing was approximately twice the HETS for extraction. The composition of the aqueous phase influenced not only the distribution coefficient but also the HETS. HETS values must be regarded as specific for given conditions of type of packing, solvent composition, aqueous phase composition, and rate of flow. The extraction of U was 99.4% complete in one test where the height of packed section was 118.5 cm. The pH of the solution was below O and the distribution coefficient was 2.03. At similar flow rates in the same apparatus a solution at pH 1.7, which had a distribution coefficient of 1.51, gave only 94.5% extraction. The thorium content of the extracts was reduced to values as low as 0.2 mg per liter in the upper section of the column. This was a reduction to $\frac{1}{350}$ of the thorium content of the extract before scrubbing. A mathematical relationship between HETS, HTU_{OE}, and HTU_{OW} was developed. Hence it is immaterial whether column design if based on HETS or HTU_{OE} or HTU_{OW}. On the whole, countercurrent extraction with dibutyl cellosolve, followed by scrubbing the extract with calcium nitrate, was found to be a practical method for separating U from Th and purifying the U to any desired degree of purity. (auth)

7541 NAA-SR-242
North American Aviation, Inc., Downey, Calif.
THE VOLATILIZATION OF TRACER PLUTONIUM FROM MOLTEN URANIUM METAL. D. Cubicciotti. Apr. 30, 1953. Decl. Mar. 2, 1957. 19p. Contract AT-11-1-GEN-8. $0.30(OTS).

The rate of volatilization of Pu from molten U in vacuum has been studied at 1495, 1600, and 1680°C. The feasibility of the large scale recovery of Pu from U by volatilization has been considered. (auth)

7542 NAA-SR-1015
North American Aviation, Inc., Downey, Calif.
SEPARATIONS CHEMISTRY QUARTERLY PROGRESS REPORT [FOR] JANUARY–MARCH 1954. E. E. Motta, D. W. Bareis, and D. Cubicciotti, eds. June 1, 1954. Decl. Mar. 12, 1957. 24p. Contract AT-11-1-GEN-8. $4.80(ph OTS); $2.70(mf OTS).

Scale-up work on high temperature fuel recovery processes has continued. Small scale experiments on high temperature methods for processing molten irradiated U fuel have been made with spent fuel slugs. The results of direct Pu evaporation, treatment with fused fluorides, and oxide scavenging were very similar to those found with tracer experiments. Small scale tracer experiments on the extraction of Pu from U with both Ca and Mg show that up to 80% of the Pu was extracted with Ca and 20% with Mg. Treatment of U amalgams with KCl–LiCl at 350°C removes rare earths to some extent. (auth)

7543 NAA-SR-1071
North American Aviation, Inc., Downey, Calif.
SEPARATIONS CHEMISTRY QUARTERLY PROGRESS REPORT, APRIL–JUNE 1954. David W. Bareis, Daniel D. Cubicciotti, and E. E. Motta, eds. Oct. 1, 1954. Decl. Mar. 2, 1957. 17p. Contract AT-11-1-GEN-8. $3.30(ph OTS); $2.40(mf OTS).

Scale-up experiments on high temperature fuel recovery processes included the dummy run phase on the handling of 1-kg samples of molten, non-irradiated uranium in the hot cell. The next step involves the use of spent X-10 fuel slugs. Small scale experiments with X-10 uranium on the extraction of Pu with Mg show that as much as 80% of the Pu can be removed in one pass. Treatment of uranium with fused fluorides can remove at least 90% of the Pu in one pass. Oxide scavenging with ZrO_2 is very effective in removing rare earths. (auth)

7544 NAA-SR-1160
North American Aviation, Inc., Downey, Calif.
SEPARATIONS CHEMISTRY QUARTERLY PROGRESS REPORT [FOR] JULY–SEPTEMBER 1954. E. E. Motta, Daniel D. Cubicciotti, and David W. Bareis, eds. Apr. 15, 1955. Decl. Feb. 27, 1957. 33p. Contract AT-11-1-GEN-8. $6.30(ph OTS); $3.00(mf OTS).

A scaled-up oxide slagging experiment involving partial decontamination of 916 gm of molten irradiated U fuel has been successfully completed. About 70% of the fission product radioactivity was removed. Small-scale Mg extraction experiments on molten U fuel have shown a constant distribution coefficient for Pu over the range of Pu concentrations from 0.03 to 97 ppm and for a U/Mg weight ratio of 1 to 3. Also, Mg has been found to be effective in removing certain fission products, namely, Cs, Sr, Te, and the rare earths. A preliminary experiment has been made to test the possibility of continuous Pu extraction by recycling the Mg over the U. The results obtained in fused fluoride salt extraction experiments are summarized and show that in several instances more than 99% of the Pu, Ce, rare earths, Sr, and Cs was removed after treatment with fused fluorides. (auth)

7545 NAA-SR-1289
North American Aviation, Inc., Downey, Calif.
SEPARATION CHEMISTRY QUARTERLY PROGRESS REPORT [FOR] OCTOBER–DECEMBER 1954. E. E. Motta, Daniel D. Cubicciotti, and David W. Bareis, eds. July 15, 1955. Decl. Feb. 28, 1957. 29p. Contract AT-11-1-GEN-8. $0.35(OTS).

Oxide scavenging and Pu volatilization experiments were carried out on irradiated U slugs. The separation of Pu and fission products from U by liquid Mg extractions and fused fluoride extractions was studied. Research was initiated on the decontamination of Th–U alloys using solid scavenger techniques. (C.W.H.)

7546 NAA-SR-1397
North American Aviation, Inc., Downey, Calif.
SEPARATIONS CHEMISTRY QUARTERLY PROGRESS REPORT FOR JANUARY–MARCH 1955. E. E. Motta, David W. Bareis, Daniel Cubicciotti, and Archie G. Buyers, eds. Oct. 15, 1955. Decl. Feb. 27, 1957. 39p. Contract AT-11-1-GEN-8. $0.40(OTS).

Experiments with U fuel have demonstrated that Ce, rare earths, Sr, Cs, Te, and Pu are efficiently removed by vaporization at 1730° to 1775°C. Treated uranium contained only negligible traces of fission products as well as 24% Pu. Oxide scavenging of U fuel has demonstrated Cs and rare earth removal to the extent of 97 to 99% with slight Pu decontamination. Laboratory scale equilibration of molten U and Mg both in batch and recycling processes has removed 50 to 96% Pu. Addition of Sn, Li, or Fe had little or no effect upon Pu distribution. Small-scale preliminary distribution data have indicated that Pu can be extracted from spent U and reduced into fresh U using molten UF_4 as transfer medium. Urania crucibles have proven more effective as oxide scavenging agents than formation of oxide skins. Ninety-nine % Sr, Cs, rare earths, and Ce were removed by UO_2 crucible scavenging. Pu and Ru remained in the U fuel. Te was volatilized. Urania and alumina crucibles were found more suitable than zirconium nitride and titanium boride for containing molten Th. Carbide scavenging of Th–U alloy has partially removed Cs, Sr, and Te. Ru and Pa remained in the alloy. Cs and Sr were completely volatilized from Th–U alloy at 1850°C. Partial decontamination of Ce and rare earths was achieved. Pa, Ru, and Zr are not removed to any great extent. The de Boer-Van Arkel iodide process has been applied to irradiated Th–U alloy. An equilibration at 1150°C of Th–U–Mn alloy with ThF_4 has been carried out. Apparatus for arc zone melting has been developed and applied to zone melting of irradiated Th–U alloy. This resulted in complete removal of Sr by volatilization. (auth)

7547 NAA-SR-1435
North American Aviation, Inc., Downey, Calif.
THE PYROCHEMICAL RECYCLING OF PLUTONIUM FROM SPENT TO FRESH URANIUM FUEL USING MOLTEN UF_4 AS TRANSFER MEDIUM. Archie G. Buyers. Oct. 1, 1955. Decl. Feb. 27, 1957. 31p. Contract AT-11-1-GEN-8. $4.80(ph OTS); $2.70(mf OTS).

The pyrochemical transfer of Pu from spent to fresh U using molten UF_4 as transfer agent is demonstrated. The resultant U–Pu alloy contained fission products in small fractions of the amounts originally present in the spent fuel. Distribution of Pu and fission products throughout recycling is correlated with thermodynamic data. Application of this method for control of Pu concentration in a Na graphite reactor fuel cycle and fast breeder reactors is suggested. (auth)

7548 NAA-SR-1515
Atomics International, North American Aviation, Inc., Canoga Park, Calif.
SEPARATIONS CHEMISTRY QUARTERLY PROGRESS REPORT FOR APRIL–JUNE 1955. E. E. Motta, A. G. Buyers, and S. J. Yosim, eds. Apr. 1, 1956. Decl. Feb. 27, 1957. 46p. Contract AT-11-1-gen-8. $0.45(OTS).

Small-scale experiments involving metal extraction, fused-salt slagging, and oxide scavenging of spent U fuel and irradiated Th–U alloys are reported. Scale-up work on various processes is described. (auth)

7549 NAA-SR-1676
Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.
SEPARATIONS CHEMISTRY QUARTERLY PROGRESS REPORT [FOR] JANUARY–MARCH 1956. E. E. Motta, A. G. Buyers, and J. R. Foltz, eds. Oct. 1, 1956. Decl. Mar. 1, 1957. 23p. Contract AT-11-1-GEN-8. $0.35(OTS).

Hot cell facilities have been relocated, modified, and improved to facilitate experimental work on oxide slagging and other pyrometallurgical fuel processing studies. Uranium slug casting and slug cropping work has suggested the possibility of shearing slugs to near finish dimensions. Design and cost estimates for the pyroprocessing and refabrication experiment are nearing completion. Research concerning small-scale high temperature processing of irradiated Th has disclosed the following information: The de Boer Iodide Process has removed all but a trace of Ru, as well as greater than 90% Sr and Cs; arc-melting has decontaminated irradiated Th alloy of ~32 to 95% of the

Ce and rare earths, 80 to 90% Cs and Sr, and 45 to 96% Te; and liquid oxide slagging, using an equimolar mixture of CaF_2–CaO, has removed 65 to 76% Ce and rare earths, 99% Sr, 74% Te, and 56% Zr. Inductively heated, non-irradiated Th–U alloy has been successfully melted through a frozen CaF_2 salt bed, which is a process obviating crucible interference in equilibration or slagging procedures. Electrorefining apparatus for application in decontamination studies of irradiated Th–U alloy, using molten salt baths, has been constructed and tested. (auth)

7550 NAA-SR-1724
Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.
PYROMETALLURGICAL PROCESSES FOR THORIUM–URANIUM FUEL. Semiannual Progress Report [for] July–December 1955. A. G. Buyers, ed. Oct. 1, 1956. Decl. Feb. 28, 1957. 29p. Contract AT-11-1-GEN-8. $0.35 (OTS).

A report of progress is made on pyrometallurgical process for Th–U fuel. Investigation of pyrochemical processing of Th–U fuels has indicated that the de Boer Iodide Refining Process offers promise for fuel decontamination by removing fission products Ru, Cs, Ce, Sr, and rare earths. A two-step process consisting of fractional volatilization of iodides and a de Boer decomposition of condensed iodides in succession, should provide for removal of the fission products mentioned above, Pa, Zr, Nb, Mo, and Te. Studies are continuing on volatilization of fission products from arc-melted Th–U alloy samples and on molten salt slagging of Th–U. Their potential value as indicated by preliminary data warrants further work. Work has been terminated on decontamination by volatilization from Th–U melts contained in a crucible, solid slagging of Th–U, and arc zone melting. These processes, while offering reasonable decontamination from Ce, Sr, Cs, Te, rare earths and in the latter case Ru, did not warrant further investigation. The results from the volatilization experiments which employed hydrided and dehydrided Th at slightly below their melting points, are sufficiently promising to require further consideration. (auth)

7551 NYO-1082
Foote Mineral Co., Philadelphia.
LOW-HAFNIUM ZIRCONIUM PROJECT PROGRESS REPORT [FOR] JANUARY 16, 1950 TO MARCH 16, 1950. S. C. Ogburn, Jr. and H. M. Fisher. Appendix: PILOT PLANT OPERATING MANUAL; LOW-HAFNIUM ZIRCONIUM OXIDE BY DISTILLATION PROCESS. H. M. Fisher. Mar. 28, 1950. Decl. Feb. 7, 1957. 31p. Contract AT(30-1)-543. $4.80(ph OTS); $2.70(mf OTS).

7552 ORNL-287
Oak Ridge National Lab., Tenn.
TECHNICAL DIVISION INTRA-DIVISIONAL MEMO–CHEMICAL DEVELOPMENT LABORATORY AND SEMI-WORKS FOR MONTH ENDING JUNE 30, 1949. F. L. Steahly. Aug. 18, 1949. Decl. Mar. 2, 1957. 47p. Contract W-7405-eng-26. $7.80(ph OTS); $3.30(mf OTS).

Progress is reported on Redox, UAP, TBP, RaLa, and 25 Processes. (F.S.)

7553 ORNL-372
Oak Ridge National Lab., Tenn.
METHYL ISOBUTYL KETONE AS A SOLVENT FOR U^{233} SEPARATION FROM THORIUM AND FISSION PRODUCTS IN CONTINUOUS COLUMN OPERATIONS. D. E. Ferguson and R. E. Lueze. June 5, 1950. Decl. Mar. 2, 1957. 12p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

The use of methyl isobutyl ketone as a solvent for U^{233} recovery from solutions of irradiated Th has been evaluated in countercurrent packed columns. (auth)

7554 ORNL-651
Oak Ridge National Lab., Tenn.
STATUS REPORT ON THE 23 PILOT PLANT PROGRAM. D. E. Ferguson, H. K. Jackson, and E. L. Nicholson. Mar. 27, 1950. Decl. Mar. 4, 1957. 39p. Contract W-7405-eng-26. $6.30(ph OTS); $3.00(mf OTS).

The results of the first 10 hot runs completed in the 23 Process Pilot Plant are presented. A program is presented for the remaining 10 runs. (W.L.H.)

7555 ORNL-874
Oak Ridge National Lab., Tenn.
EVALUATION OF ALTERNATE SOLVENTS FOR DIETHYL ETHER IN THE EXTRACTION OF URANIUM FROM MALLINCKRODT TYPE ORE FEEDS. T. C. Runion. Nov. 6, 1950. Decl. Mar. 5, 1957. 23p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

An evaluation of solvents for use in the extraction of uranium from Mallinckrodt-type ore feeds is presented. Of the solvents examined, namely, tributyl phosphate (TBP), diethyl ether, methyl isobutyl ketone (hexone), dibutoxytetraethyleneglycol (pentaether), diisopropyl ether, and dibutyl cellosolve, only TBP shows feasibility of replacing diethyl ether with a minimum of short-range development. (F.S.)

7556 ORNL-1148
Oak Ridge National Lab., Tenn.
LABORATORY DEVELOPMENT OF THE MTR-RaLa PROCESS FOR THE PRODUCTION OF BARIUM^{140}. R. E. Blanco. Aug. 19, 1952. Decl. Mar. 2, 1957. 70p. Contract W-7405-eng-26. $10.80(ph OTS); $3.90(mf OTS).

7557 ORNL-1224
Oak Ridge National Lab., Tenn.
CONTINUOUS SOLVENT RECOVERY PROCESS USING PULSE CONTACTING COLUMN. C. V. Ellison. Apr. 1, 1952. Decl. Mar. 5, 1957. 57p. Contract W-7405-eng-26. $9.30(ph OTS); $3.60(mf OTS).

This report summarizes the progress in the laboratory development of a continuous solvent recovery process for the tributyl phosphate–diluent mixture used in the Purex Process. Investigations were made with 0.5-in.-diameter pulse columns, using a nine-foot sodium carbonate contacting column and a four foot water wash column. (auth)

7558 ORNL-1690
Oak Ridge National Lab., Tenn.
SEPARATION OF URANIUM(VI) FROM OTHER METALS BY FILTER PAPER PARTITION CHROMATOGRAPHY WITH 2-METHYLTETRAHYDROFURAN AND ITS QUANTITATIVE DETERMINATION. I. SEPARATION STUDIES BY USE OF RADIOISOTOPES. H. P. Raaen and P. F. Thomason. Aug. 4, 1954. Decl. Mar. 2, 1957. 48p. Contract W-7405-eng-26. $0.45(OTS).

Radioisotopes were used in a study of the microseparation of U^{6+} from other metals by filter-paper partition chromatography with 2-methyltetrahydrofuran. Autoradiography of the chromatograms established the efficiency of the separations. The sorption gradient of the separated U^{233} was determined by α-count measurements of the chromatograms; the quantitative removal of U^{233} from the paper strip was demonstrated by the same technique. Maximum condensation of the U^{233} band was effected by a 1.75-hr elution with H_2O-saturated (at 25°C) 2-methyltetrahydrofuran that contained 2.5% (v/v) HNO_3. The average R_f value for U^{233} was 0.95. Of the radioisotopes studied, Ru^{106}, Rh^{106}, W^{185}, and Hg^{203} were not separated from U^{233}. The results for Sn^{113} and Sb^{124} were not conclusive. The behavior of Hg^{203} was similar to that of U^{233}. The α-count data are presented graphically, and positive prints of the autoradiograms are shown. (auth)

7559 Y-479

Carbide and Carbon Chemicals Corp. Y-12 Plant, Oak Ridge, Tenn.

THE USE OF BORON AS A COMPLEXING AGENT FOR FLUORIDE IN THE RECOVERY OF URANIUM AND THE SEPARATION OF BORON FROM URANIUM. J. M. Googin and T. P. Sprague. Aug. 25, 1949. Decl. Feb. 11, 1957. 7p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

The use of borax in place of aluminum nitrate for the leaching of K-25 trap carbons would be much more economical since borax is much cheaper per pound and requires only one-fourth as much. Three percent more of the uranium is leached from the pellets with the borax–nitric acid mixture than with aluminum nitrate–nitric acid using a solution which is 3M HNO_3 and having one mole of boron or aluminum to complex two moles of fluoride. The corrosion tests on stainless steel type 316 show only fifty percent as much corrosion at 80°C with the borax mixture as with aluminum nitrate. The boron can be separated quantitatively from uranium with a single peroxide precipitation. If necessary, the remaining uranium could be recovered from the peroxide effluent after decomposing the excess H_2O_2 by either a NaOH or NH_4OH precipitation and at the same time effect some separation of the uranium and boron. This material could then be dissolved and reprecipitated with hydrogen peroxide. (auth)

7560 Y-838

Carbide and Carbon Chemicals Co. Y-12 Plant, Oak Ridge, Tenn.

CRITICAL LITERATURE SURVEY OF TRIBUTYL PHOSPHATE AS A URANIUM EXTRACTANT. W. B. Wright, Jr. Jan. 14, 1952. Decl. Feb. 11, 1957. 27p. Contract W-7405-eng-26. $4.80(ph OTS); $2.70(mf OTS).

Important data and discussions obtained from ten primary references pertaining to the solvent extraction of U with TBP are collected in this survey. Topics covered are: physical and chemical properties of pure and diluted TBP, organic diluent for TBP, the mechanism of U extraction by TBP, factors influencing the extraction, interferences and other ions extracted, recovery of U from the organic phase, and purification of TBP for re-use. (auth)

Refer also to abstracts 7805 and 7838.

SYNTHESES

7561 A-2122

Columbia Univ., New York. Div. of War Research.

CONVERSION OF BF_3 TO BCl_3. Clyde A. Hutchison and Richard K. Eberts. Jan. 25, 1945. Decl. Feb. 12, 1957. 111p. Contract W-7405-eng-50. (100B-R-283). $21.30 (ph OTS); $6.90(mf OTS).

A method for the preparation of pure crystalline B has been developed. The third step, the $AlCl_3$ conversion by bomb, packed tubes, stirred vessels, and the recovery from residues is reported. Analytical procedures for separation of BF_3 and BCl_3 by distillation and the viscosity of BF_3 are included. (F.S.)

TRANSURANIC ELEMENTS AND COMPOUNDS

7562 HW-17265

Hanford Works, Richland, Wash.

THE PREPARATION OF THIN FILMS OF PLUTONIUM BY ELECTRODEPOSITION. H. W. Miller and R. J. Brouns. Mar. 22, 1950. Decl. Feb. 19, 1957. 10p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

An electrodeposition procedure was found most satisfactory for the preparation of thin and uniform films of plutonium. The procedure consists of the electrolytic reduction of plutonyl ion to Pu(IV) in a 1N KOH solution. The Pu(IV) precipitates as the hydrous hydroxide on a platinum cathode disk and is subsequently ignited to the oxide. Impurities which precipitate in an alkaline solution and carry or adsorb plutonium must be absent. Lanthanum, chromium, and silver have been shown to interfere and probably aluminum, iron, and all 4 f and 5 f rare earths would likewise cause difficulty. Other factors which cause low yields are incomplete oxidation of plutonium to plutonyl, and low alkalinity. Yields of 95 to 99% were obtained from solutions containing up to 7 micrograms of plutonium in 10 ml of a 1N KOH solution using well-polished platinum cathodes, 500 rpm stirring, and a two hour plating time at room temperature. (auth)

7563 HW-18188

Hanford Works, Richland, Wash.

THE SOLUBILITY OF PLUTONIUM(IV) MONOBUTYL PHOSPHATE. Robert Lee Moore. June 28, 1950. Decl. Feb. 19, 1957. 15p. Contract W-31-109-eng-52. $3.30 (ph OTS); $2.40(mf OTS).

Plutonium (IV) monobutyl phosphate was prepared. Chemical analysis indicates the composition $Pu(C_4H_9PO_4)_2 \cdot xH_2O$. The solubility of this material was investigated as a function of time, temperature, acidity, concentration of MBP and the presence of dibutyl phosphate. MBP was found to precipitate Pu (IV) from a mixture of the (III) and (IV) states in 3 M HNO_3. Zirconium monobutyl phosphate was found to precipitate Pu (IV) from tracer solutions suggesting its analytical application to valence state determinations. (auth)

URANIUM AND URANIUM COMPOUNDS

7564 A-1038

[Madison Square Area, Manhattan District, N. Y.]

EXCESS OF MAGNESIUM USED AS REDUCTANT IN PRODUCTION. W. H. Keller. June 20, 1944. Decl. Feb. 12, 1957. 10p. $3.30(ph OTS); $2.40(mf OTS).

Results indicate that optimum yields of uranium are obtained from the reduction of UF_4 when excess Mg (~7.5%) is used. Operating conditions include a UF_4 charge of 55 lb and furnace temperatures of 1200°F. (C.W.H.)

7565 A-1053

Du Pont de Nemours (E. I.) and Co. Engineering Dept., Wilmington, Del.

PREDICTION OF MAGNESIUM BEHAVIOR IN THE BISCUIT REACTION. Aug. 14, 1944. Decl. Feb. 14, 1957. 28p. $4.80(ph OTS); $2.70(mf OTS).

The rates at which magnesium samples evolve H_2 from acid solutions were investigated. Results indicate that a definite relationship exists between the U biscuit yield and rate of hydrogen evolution. (C.W.H.)

7566 A-3789(Sect. 5)

Du Pont de Nemours (E. I.) & Co. TNX Div., Wilmington, Del.

CLINTON LABORATORIES PROCESS MANUAL. SECTION 5. METAL SOLUTION. W. E. Kirst. [1943]. Decl. Feb. 12, 1957. $6.30(ph OTS); $3.00(mf OTS).

A detailed description of the process for dissolution of jackets and slugs is given. The plant and its operation are described and flowsheets and drawings included. (T.R.H.)

7567 ANL-4085

Argonne National Lab., Lemont, Ill.

ASPECTS OF LARGE-SCALE PRODUCTION OF U^{233}

Leonard I. Katzin. Dec. 1947. Decl. Feb. 13, 1957. 44p. Contract W-31-109-eng-38. $7.80(ph OTS); $3.30(mf OTS).

The methods and problems of the production of U^{233} are surveyed. (B.J.H.)

7568 ANL-5102
Argonne National Lab., Lemont, Ill.
THE ELECTROLYTIC REFINING OF URANIUM. C. Marzano and R. A. Noland. Aug. 2, 1953. Decl. Feb. 16, 1957. 60p. Contract W-31-109-eng-38. $0.50(OTS).

The electrolytic refining of natural U in fused-salt baths composed of various eutectics of alkali metal chlorides into which were dissolved UF_4 or UCl_3 is described. Near spectroscopic purity was achieved with two electrolytes. One electrolyte was composed of 33 wt. % UCl_3, 20% KCl, and 47% $BaCl_2$ and was maintained molten at 600 to 625°C. The other electrolyte containing 33 wt. % UCl_3 or UF_4, 30% LiCl and 37% KCl melted at a lower temperature, permitting operation at 390 to 425°C. The first electrolyte was shown to be more sensitive to impurities in the feed salts; the second was rather tolerant to impurities, permitting lower purity salts and metal to be used without lowering the high purity of the refined product. Other electrolytes melting at higher temperatures were also investigated. The metal to be refined was made the anode, and the product, a woven mass of dendrites and small crystals, was collected on a Mo or Ta cathode. Argon or He was required as a cover gas to protect the electrolyte and to protect the U crystals after removal from the bath. Samples of the refined material were vacuum-melted in UO_2 crucibles without significant change in purity. The resultant metal was soft, ductile, and easily worked. (auth)

7569 BMI-1013
Battelle Memorial Inst., Columbus, Ohio.
GRAIN REFINEMENT OF THE URANIUM-5 w/o CHROMIUM ALLOY BY TERNARY ADDITIONS. Henry A. Saller, Ronald F. Dickerson, William E. Murr, and Arthur A. Bauer. July 5, 1955. Decl. Feb. 13, 1957. 12p. Supported by Atomic Power Development Associates through Contract W-7405-Eng-92. $0.25(OTS).

Refinement of grain size of the base uranium–chromium eutectic alloy (uranium–5 wt. % chromium) by ternary additions is discussed. Thirteen additions to the base alloy were investigated, and three additions were found to exhibit worthwhile grain-refinement tendencies. The three alloys, uranium–5 wt. % chromium–0.2 wt. % molybdenum, uranium–5 wt. % chromium–0.2 wt. % germanium, and uranium–5 wt. % chromium–0.2 wt. % niobium, were heat treated to produce final grain-size structures of between 0.015 and 0.033 mm. (auth)

7570 BMI-1103
Battelle Memorial Inst., Columbus, Ohio.
DIFFERENTIAL THERMAL ANALYSIS OF URANIUM TETRAFLUORIDE–URANIUM DIOXIDE MIXTURES. Robert A. Ewing and Arthur E. Bearse. June 28, 1956. Decl. Feb. 20, 1957. 22p. Contract W-7405-eng-92. $0.35(OTS).

Approximate melting points have been determined for 5 samples of UF_4 representing incompletely converted UO_2 and covering the range from about 2 to 20 wt.% UO_2, using the method of differential thermal analysis. The results indicate that melting temperatures are in the range of 910 to 940°C. No significant correlation between melting point and UO_2 content was observed, possibly because of eutectic formation. Similar results were obtained on synthetic mixtures of UF_4 containing from 10 to 40 wt.% UO_2. (auth)

7571 BMI-JDS-227
Battelle Memorial Inst., Columbus, Ohio.
RECOVERY OF THORIUM AND URANIUM FROM MONAZITE SANDS. Progress Report for the Month of February 1950. Arthur E. Bearse. Feb. 28, 1950. Decl. Feb. 12, 1957. 31p. Contract AT-30-1-Gen-228. $4.80(ph OTS); $2.70(mf OTS).

Laboratory and pilot-plant studies were continued on ore dressing for Ti removal, acid dissolution of hydrous oxide cake (from reaction of monazite with caustic soda), purification and separation of Th and U from oxide cake solvent extraction with TBP, and separation of U from Th by anion exchange. (C.W.H.)

7572 CF-51-10-91
Oak Ridge National Lab., Tenn.
U–AL ALLOY DISSOLVING IN HNO_3. Floyd L. Culler. Oct. 15, 1951. Decl. Feb. 13, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The hydrogen evolution during the nitric acid dissolution of Al–U alloys and the corrosion of stainless steel by nitric acid containing Hg salts were investigated. (C.W.H.)

7573 CF-54-12-169
Oak Ridge National Lab., Tenn.
EXCER: MOVING-BED REACTOR STUDIES. A LITERATURE SURVEY CONCERNING THE ONE-STEP PRODUCTION OF UF_4 FROM UO_3. J. E. Moore. Dec. 9, 1954. Decl. Feb. 16, 1957. 31p. Contract W-7405-eng-26. $6.30(ph OTS); $3.00(mf OTS).

A literature survey of project literature was made in connection with the moving-bed reactor study of the production of UO_3 by simultaneous dehydration, reduction, and hydrochlorination of UF_3 with HF and NH_3. UO_3 or UO_2 hydrate can be completely converted to UF_4 and the simultaneous reduction-hydrofluorination reactor is potentially capable of converting by-product UO_2F_2 to UF_4. (F.S.)

7574 CF-56-6-125
Oak Ridge National Lab., Tenn.
DEHYDRATION OF URANIUM TETRAFLUORIDE HYDRATE. L. L. Annas. June 22, 1956. Decl. Feb. 14, 1957. 7p. Contract [W-7405-eng-26]. $0.25(OTS).

Results indicate that $UF_4 \cdot {}^3/_4\ H_2O$ can be successfully dehydrated continuously and batchwise in a dry N_2 atmosphere at elevated temperatures. (T.R.H.)

7575 CN-1206
Clinton Labs., Oak Ridge, Tenn.
ENGINEERING STUDIES OF DRY FLUORIDE PROCESS. D. S. Webster. Jan. 10, 1944. Decl. Feb. 14, 1957. 17p. Contract [W-7405-eng-39]. (A-1686). $3.30(ph OTS); $2.40(mf OTS).

7576 CT-2733
National Bureau of Standards, Washington, D. C.
FINAL REPORT ON THE RESULTS OF EXPOSURE OF TUBALLOY [URANIUM] AT ELEVATED TEMPERATURE IN AIR, NITROGEN, AND NEUTRAL ATMOSPHERES. W. E. Lindlief and V. C. F. Holm. Feb. 10, 1945. Decl. Feb. 20, 1957. 17p. Contract [W-7401-eng-37]. $6.30(ph OTS); $3.00(mf OTS).

Uranium oxides UO_3 and U_3O_8 were the products of air oxidation of U at temperatures between 100 and 500°C. The oxides formed a protective coating at temperatures of 200°C or lower. Nitrogen reacts with U at temperatures up to 850°C to form the nitride, $UN_{1.8}$. Two physically distinct forms of the nitride which was rapidly oxidized to U_3O_8 on exposure to air at high temperatures were formed. Uranium was annealed in the α and β ranges in purified A and He and in high vacuum; no true anneal was observed in γ range. (C.W.H.)

7577 FMPC-122

National Lead Co. of Ohio, Cincinnati.

STANDARD OPERATING PROCEDURE FOR PRODUCTION ORDER NO. 17. [1953?]. Decl. Feb. 23, 1957. 17p. Contract AT(30-1)-1156. $0.30(OTS).

Procedures are given for straightening, sawing, and grinding enriched rods 0.6 in. in diameter and 4 ft long. Criticality, health and safety, accountability, and enriched material identification measures are outlined. (J.E.D.)

7578 FMPC-176

National Lead Co. of Ohio, Cincinnati.

PILOT PLANT SCALE HYDROGEN REDUCTION OF URANIUM HEXAFLUORIDE TO URANIUM TETRAFLUORIDE. R. M. Spenceley. T. C. Runion, ed. Mar 30, 1953. Decl. Feb. 16, 1957. 26p. Contract AT(30-1)-1156. $4.80(ph OTS); $2.70(mf OTS).

The FMPC pilot plant UF_6 to UF_4 reduction unit was designed to provide data for larger production units and to fulfill limited production orders for UF_4. (auth)

7579 FMPC-433

National Lead Co. of Ohio., Cincinnati.

MANUAL OF QUALITY CHARACTERISTICS OF INTERMEDIATE URANIUM FUEL ELEMENT PRODUCTS. R. A. Matuszeski. June 7, 1954. Decl. Feb. 19, 1957. 49p. Contract AT(30-1)-1156. $0.45(OTS).

Common surface defects and internal defects of U derbies, ingots, billets, rods, and slugs are illustrated photographically. The defects shown are those visible to the naked eye or at low magnification. Clean derbies, ingots, and slugs are also illustrated. (auth)

7580 HW-18492

Hanford Works, Richland, Wash.

SETTLING AND DISSOLUTION CHARACTERISTICS AND COMPOSITION OF HANFORD WASTE METAL SLUDGE. H. L. Brandt and W. A. Burns. June 23, 1950. Decl. Feb. 20, 1957. 16p. Contract W-31-109-Eng-52. $3.30 (ph OTS); $2.40(mf OTS).

Qualitative observations and chemical analysis of 4-year old Hanford waste metal sludge are presented. Settling rates of sludge–supernatant slurries were measured. It was proposed that sludge and supernatant be slurried by high-pressure jets of supernatant. (T.R.H.)

7581 HW-20598

Hanford Works, Richland, Wash.

EFFECT OF FURNACE PRESSURES ON QUALITY OF REMELTED URANIUM. Final Report, Production Test 314-59-M. W. T. Kattner. Feb. 15, 1951. Decl. Feb. 19, 1957. 10p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

The effect of melt-plant casting pressures upon metal quality was evaluated using castings poured from charges containing either unpickled or pickled briquettes. The billets from the charges containing pickled briquettes were superior with respect to melt-plant yield, reactivity, nitrogen content, and the frequency of tears occurring in the rods during rolling. The lower pressures were observed to be beneficial in reducing the frequency of tears in rolled rods. The fact that no other evidence of significant difference in metal quality accompanied variations in casting pressure indicates either that the casting pressure is of negligible importance or that the amount of oxides and nitrides present in the charges initially varied so greatly that the effect of pressures was lost. The latter view seems more reasonable in view of the evidence obtained. (auth)

7582 MCW-1

Mallinckrodt Chemical Works, St. Louis.

THE PREPARATION OF PURE URANYL NITRATE FROM BLACK OXIDE. Charles R. Conrad. April 1, 1946. Decl. Feb. 26, 1957. 39p. Contract W-7405-eng-1. $0.40(OTS).

Studies which led to the following simplified procedure are described: Add the finely powdered black oxide to warm diluted nitric acid until the nitric acid is completely used up, as determined by a pH test. Filter and concentrate to drive off the water not required as hexahydrate, as determined by the boiling point. Add the liquid hexahydrate to cold washed ether in an extractor fitted with a condenser. Remove the aqueous layer and wash the ether layer batchwise and counter-currently with saturated wash liquors from previous runs followed by one final distilled water wash. Extract the purified uranyl nitrate from the ether layer with a few relatively large volumes of distilled water. Boil off the ether and evaporate the aqueous solution to a boiling point of 118 to 120° to obtain liquid uranyl nitrate hexahydrate. The hazards involved and materials suitable for fabrication of full scale equipment were investigated. (J.E.D.)

7583 MITG-A39

Massachusetts Inst. of Tech., Watertown, Mass. Mineral Engineering Lab.

PRELIMINARY EXPERIMENTS IN GAMMA-COUNTING FLOTATION PULPS. James H. Pannell. Jan. 25, 1949. Decl. Feb. 27, 1957. 21p. Contract W-7405-Eng-85. $3.30(ph OTS); $2.40(mf OTS).

Gamma and beta counters were considered for continuous radioassay of plant ore pulps. Gamma counters were more rugged but less efficient than the beta counters. This low efficiency was compensated for by using a large sample to take advantage of the greater range of gamma radiation. Empirical absorption coefficients were determined and used to indicate the effect of tank size on measured activities. (auth)

7584 MITG-A88

Massachusetts Inst. of Tech., Watertown, Mass. Mineral Engineering Lab.

PRODUCTION OF CONCENTRATED LEACH SOLUTION. D'Arcy R. George. Oct. 28, 1949. Decl. Mar. 7, 1957. 15p. Contract W-7405-Eng-85. $3.30(ph OTS); $2.40 (mf OTS).

To obtain a leach solution from ores assaying 1 g/l U_3O_8, a cyclic leaching method was devised in which only part of the pregnant solution was withdrawn and the balance was recirculated for leaching fresh ore. The wash water was kept separate, its uranium precipitated with magnesia, and this precipitate returned to the leaching step. The process was tested successfully except for some difficulty in filtration. No advantage was shown for the concentrated solution in subsequent precipitation tests. The process is not recommended. (auth)

7585 MonN-268

Clinton Labs., Oak Ridge, Tenn.

STABILITY OF UF_6 IN A PILE. D. E. Hull. Mar. 4, 1947. Decl. Feb. 28, 1957. 12p. Contract W-7405-eng-39. $3.30(ph OTS); $2.40(mf OTS).

Calculations indicated that the radiolysis of UF_6 under reactor conditions should be negligible. ($2UF_6 \rightarrow 2UF_5 + F_2$). (C.W.H.)

7586 NYO-1321

Mallinckrodt Chemical Works, St. Louis.

FINAL REPORT OF PILOT PLANT WORK DONE ON THE REVERSION OF UF_4 TO UO_2 OR TO U_3O_8 BY CONTINUOUS METHODS. Period Covered May 26, 1950 to June 27, 1950. E. K. Teter. Mar. 8, 1951. Decl. Mar. 2, 1957. 10p. $0.25(OTS).

Uranium tetrafluoride can be successfully reverted to UO_2 with existing pilot plant equipment. The feed rate of dry UF_4 should not exceed 50.0 lb/hr, the retention time in the reactor should be at least $2\frac{1}{2}$ hr, the average

reactor temperature should be no lower than 1130°F, the steam rate should be in excess to force the reaction to completion, and the purging gas should be inert. Reverting to U_3O_8, using air as a purging gas was twice as slow as the reverting to UO_2, using N_2 as a purging gas, all other conditions being equal. (auth)

7587 NYO-1335
Mallinckrodt Chemical Works, St. Louis.
URANIUM METAL BY BOMB REDUCTION. V. THE REACTION OF UF_4 AND MAGNESIUM VAPOR. R. M. Paine, A. E. Ruehle, and G. W. Lewis, Jr. June 20, 1952. Decl. Mar. 2, 1957. 19p. Contract W-14-108-eng-8. $3.30(ph OTS); $2.40(mf OTS).

The reaction between UF_4 and Mg vapor was studied in the temperature range 500 to 800°C, to obtain information about possible intermediate compounds and relative rates of reaction. The reduction proceeds to UF_3, the only intermediate detected, with relative rapidity; thence to U metal, with the UF_3 concentration being the rate-determining factor in the second reduction step. At the lower temperatures it is possible to obtain total reduction to UF_3 with very little formation of U metal. Uranium trifluoride was found to be fairly stable in air at room temperature. (auth)

7588 Y-350
Carbide and Carbon Chemicals Corp. Y-12 Plant, Oak Ridge, Tenn.
PILOT SCALE TESTS OF THE PREPARATION OF URANIUM TETRAFLUORIDE FROM URANYL AMMONIUM PHOSPHATE. Don Phillips, H. M. McLeod, Jr., and J. S. Reece. Mar. 29, 1949. Decl. Feb. 7, 1957. 18p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Exploratory work was carried out to determine the feasibility of preparing uranium tetrafluoride from uranyl ammonium phosphate in the following steps: (a) calcining the UAP to remove both water and ammonia, (b) reduction of the calcined material with hydrogen to form uranous pyrophosphate, and (c) hydrofluorination of uranous pyrophosphate to form uranium tetrafluoride. Both calcination and reduction tests were carried out on a pilot scale under conditions simulating continuous operation. Attempts to produce relatively pure UF_4 on a pilot scale were not successful because the product became contaminated with scale; consequently, the reaction was carried out in laboratory equipment on a batch basis. The results of the tests showed that the process was feasible and no operational difficulties were encountered that do not appear to be amenable through the use of properly designed equipment. (auth)

7589 Y-B31-84
Oak Ridge National Lab., Y-12 Area, Tenn.
NOTES ON A METHOD OF DETERMINING THE URANIUM CONTENT OF URANIUM TETRAFLUORIDE. L. J. Brady. Aug. 22, 1952. Decl. Mar. 7, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The volumetric method of determining uranium in the tetrafluoride salt by titration with ceric sulfate after complexing the fluoride ions with aluminum ions is fast, simple, accurate, and the reproducibility of results is excellent. Uranium metal may be used as a standard for routine analysis by this procedure. The method suggested for determining the end-point is an improvement which makes for greater precision of the method. (auth)

Refer also to abstract 7498.

WASTE DISPOSAL

7590 CF-52-1-22
Oak Ridge National Lab., Tenn.
MULTI-STAGE EVAPORATION OF PUREX ACID WASTES. V. J. Reilly. Jan. 4, 1952. Decl. Feb. 14, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40 (mf OTS).

7591 CF-52-6-162
Oak Ridge National Lab., Tenn.
RESOLUTION OF THE SODIUM DIURANATE PRECIPITATE IN THE W-7 METAL WASTE TANK. J. E. Savolainen. June 26, 1952. Decl. Feb. 13, 1957. 3p. $1.80 (ph OTS); $1.80(mf OTS).

Uranium complexing with $(NH_4)_2CO_3$ and precipitate dissolving with HNO_3 were studied to determine how to dissolve the Na diuranate precipitate. $(NH_4)_2CO_3$ proceeds slowly even with agitation, and this method is not feasible. HNO_3 dissolution proceeds rapidly but attacks concrete and is not recommended because of the high cost of replacing radiochemical waste storage tanks. Chemical methods of removing sludge are not practical. (F.S.)

7592 HW-19933
Hanford Works, Richland, Wash.
VISCOSITY AND SETTLING RATES OF TBP WASTE METAL SLURRIES AS A FUNCTION OF SHEAR. L. P. Varga and M. K. Harmon. Jan. 2, 1951. Decl. Feb. 25, 1957. 41p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

The settling rate and viscosity of Hanford waste metal slurry as a function of prolonged agitation and shear were compared with the settling rate and viscosity of a synthetic slurry. The settling rate of the synthetic slurry increased sharply in the first hour of agitation, dropped slowly, but with wide fluctuation during the next four hour's agitation, and then remained constant for the duration of the experiment. The initial settling rate of the Hanford "hot" slurry dropped rapidly during the first hour's stirring, and gradually declined for an additional 20 hours before it leveled off. The viscosity of the synthetic slurry changed little with agitation. The fiscosity of the Hanford slurry rose sharply after little more than one hour's agitation time, then increased gradually, leveling off after 10 hours. (D.E.B.)

7593 KAPL-649
Knolls Atomic Power Lab., Schenectady, N. Y.
WASTE DISPOSAL PROGRESS REPORT [FOR] SEPTEMBER, OCTOBER, NOVEMBER 1951. Decl. Feb. 26, 1957. 27p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

Experimental work is reported on a pilot incinerator for contaminated combustible solids. Design basis for a production-type incinerator is summarized. The concentration of liquid waste by evaporation is reported. Data are included on evaporation equipment and tests. Factors affecting the decontamination of nitric acid by distillation are discussed. The problem of high-level waste storage at KAPL is outlined briefly. (C.H.)

7594 MLM-614
Mound Lab., Miamisburg, Ohio.
LIQUID WASTE DISPOSAL RESEARCH REPORT FOR PERIOD JUNE 4, 1951 TO SEPTEMBER 3, 1951. Oct. 8, 1951. Decl. Feb. 26, 1957. 19p. Contract AT-33-1-Gen-53. $3.30(ph OTS); $2.40(mf OTS).

Development of the ferrous sulfide, ferrous hydroxide-calcium phosphate precipitation process for decontaminating Tributyl Phosphate (TBP) Metal Recovery Wastes has been continued. Emphasis has been placed on reducing the burial volume of precipitates by eliminating addition of calcium after precipitation of ferrous sulfide and by recycling all iron except that going to burial with the first ferrous sulfide precipitate. A tentative flowsheet for the process, listing relative volumes and chemicals required, has been prepared. (auth)

7595 NYO-1580
Harvard Univ., Boston. School of Public Health.
A PRELIMINARY FIELD SURVEY OF AIR CLEANING ACTIVITIES AT ATOMIC ENERGY COMMISSION FACILITIES, SEPTEMBER–NOVEMBER 1950. August T. Rossano, Jr., Leslie Silverman, and Melvin W. First. Feb. 1, 1951. Decl. Feb. 28, 1957. 51p. Contract AT(30-1)-841. $0.45(OTS).

During the period September 6 to November 9, 1950, members of the staff of the Harvard University School of Public Health visited eleven principal Atomic Energy Commission research and production sites. The purpose of these visits was to observe air and gas cleaning installations and to consult with plant officials concerning these activities. This report contains information on air and gas cleaning methods and equipment now in use and their performance, where available, at certain sites. Pertinent comments and recommendations are included on present and proposed installations. (auth)

7596 ORNL-724
Oak Ridge National Lab., Tenn.
PHYSICAL PROPERTIES OF HANFORD METAL WASTE. C. E. Schilling. June 29, 1950. Decl. Mar. 5, 1957. 32p. Contract W-7405-eng-26. $6.30(ph OTS); $3.00(mf OTS).

The Hanford metal wastes were divided into four categories: supernate–the liquid waste; hard sludge–dense agglomerates of poorly defined crystalline carbonates approximating the hardness of soft blackboard chalk; soft sludge–an easily slurried semi-solid consisting chiefly of needle-like phosphates; and recombined sludge–a representative sample of the solid wastes as received from Hanford, shown to be a mixture of hard and soft sludges in the ratio $\frac{2}{3}$ by weight. The density of supernate, in the temperature range 24 to 74°C, varied from 1.130 to 1.103 g/ml. Hard sludge density averaged 3.0 g/ml and that of soft sludge averaged 1.84 g/ml. The consistency, or apparent viscosity, as a function of temperature, shear rate, and solids content was measured individually on slurries of recombined, soft, and hard sludges using supernate as the suspending medium. Settling rates were also run on these 3 slurries as a function of solids content. (auth)

ENGINEERING

HEAT TRANSFER AND FLUID FLOW

7597 BNL-2221
Brookhaven National Lab., Upton, N. Y.
BOILING BURNOUT NEWSLETTER NO. 3. (Classified Section). Corwin L. Rickard. Feb. 21, 1955. Decl. Feb. 16, 1957. 3p. $1.80(ph OTS); $1.80(mf OTS).

Brief data are presented on subcooled boiling burnout in a vertical Zircaloy-2 channel at exit water pressure of 2000 psia. (D.E.B.)

7598 CF-55-5-33
Oak Ridge National Lab., Tenn.
A POSSIBLE CHARACTERIZATION NUMBER FOR THORIUM OXIDE SLURRIES. P. R. Crowley. May 5, 1955. Decl. Feb. 14, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Flow properties of pumped and violently agitated thorium oxide slurries were surveyed at room temperature using a pipe line type viscometer. Analysis of the data shows a correlation between slurry concentration and yield shear stress which may be useful in characterizing slurries. (B.J.H.)

PUMPS

7599 NAA-SR-231
North American Aviation, Inc., Downey, Calif.
HIGH TEMPERATURE MATERIALS STUDIES SEMI-ANNUAL PROGRESS REPORT JULY–DECEMBER 1952. R. Keen and R. Cygan. G. M. Inman, ed. May 25, 1953. Decl. Mar. 4, 1957. 25p. Contract AT-11-1-Gen-8. $4.80(ph OTS); $2.70(mf OTS).

Studies on the use of graphite as a material of construction for liquid metal coolant circuits at temperatures above 1500°C are presented. Methods of fabricating intricate parts and methods of connecting parts for fusion joints have been developed. A small pump loop made entirely of graphite was constructed and operated for 60 hr using liquid tin as a coolant. A pump loop was fabricated and operated for approximately 500 hr to evaluate component designs and system characteristics. A constant flow of 0.82 gpm of the Pb–Bi eutectic was maintained. (C.H.)

MINERALOGY, METALLURGY, AND CERAMICS

CERAMICS AND REFRACTORIES

7600 A-1036
Iowa State Coll., Ames.
COMPARISON OF REFRACTORIES AS BOMB LINING MATERIALS IN PRODUCTION OF METAL. W. H. Keller. June 11, 1944. Decl. Feb. 7, 1957. 106p. $15.30(ph OTS); $5.40(mf OTS). $16.80(ph OTS); $5.70(mf OTS).

The refractory materials (fused magnesium oxide, fused dolomitic oxide, and calcium lime) were evaluated as bomb-lining material for U production. Calcium lime appeared to be the most economical and productive type of liner studied. Recommendations for improving the lime quality are presented. (C.W.H.)

7601 BMI-1117
Battelle Memorial Inst., Columbus, Ohio.
EFFECT OF CERAMIC OR METAL ADDITIVES IN HIGH-UO_2 BODIES. Donald J. Bowers, Walter A. Hedden, M. Jack Snyder, and Winston H. Duckworth. July 24, 1956. Decl. Feb. 13, 1957. 28p. Contract W-7405-eng-92. $0.35(OTS).

Improvements in corrosion and thermal-fracture resistance of a UO_2 ceramic were sought through the use of additions of BeO, Be, CeO_2, SiO_2, Si, Si_3N_4, or ZrO_2. Nuclear considerations dictated that the modified bodies contain at least 80 volume per cent of UO_2 after sintering. Sound specimens meeting this UO_2-content requirement were made with the BeO, CeO_2, SiO_2, and ZrO_2 additions. These ceramics, with the exception of UO_2-SiO_2, had as good resistance to 650 F water and 750 F steam as the unadulterated UO_2 ceramic. Adding refractory-grade BeO to the UO_2 powder compacts improved the thermal-fracture resistance of sintered specimens by a factor of about 1.7. Readily sinterable BeO powder did not give this benefit. Variations in sintering treatment appreciably affected the thermal-fracture characteristics of the UO_2–BeO specimens. Minor additions of titania or ceria to UO_2-BeO powder compacts resulted in increased densities at lower sintering temperatures, but thermal-fracture resistance was not improved through their use. Dense, strong specimens were made using CeO_2 as the additive to UO_2. CeO_2 was the most effective density promoter during sintering of the major additives studied. UO_2–CeO_2 specimens, however, were only slightly better in thermal-fracture resist-

ance than those of UO_2 alone. ZrO_2 and UO_2 combinations had adequate density and strength, but the ZrO_2 had a deleterious effect on the thermal-fracture resistance of UO_2. UO_2–Si cermet specimens had low density and strength, presumably because of the formation of a reaction product. The reaction occurred on sintering in H, A, or in a vacuum, and, to a lesser degree, on hot pressing. The reaction product was isolated, but not identified. It was unstable when resintered by itself and with UO_2. Satisfactorily dense and sound sintered compacts were not produced with either Be or Si_3N_4 as the additive. (auth)

Refer also to abstract 7898.

CORROSION

7602 ANL-4973
Argonne National Lab., Lemont, Ill.
AUTOCLAVE TESTING OF MECHANICALLY JACKETED THORIUM SLUGS IN WATER. Frank L. Yaggee. Jan. 29, 1953. Decl. Feb. 12, 1957. 19p. Contract W-31-109-eng-38. $0.30(OTS).

Water corrosion tests on mechanically jacketed and pinholed Th slugs show that these slugs fail in a manner similar to that observed for mechanically jacketed and tested U slugs. The proposed mechanism for the H_2O corrosion of these jacketed slugs is analogous to the H_2O corrosion mechanism of jacketed U slugs presented in the project literature. A bare Th slug appeared to be more resistant to corrosion by H_2O than a mechanically jacketed slug during the first half of the autoclave test. After approximately 90 hr of testing both the bare and the mechanically jacketed Th slugs were severely corroded by H_2O. (auth)

7603 BMI-927
Battelle Memorial Inst., Columbus, Ohio.
CORROSION OF TYPE 347 STAINLESS STEEL IN THE URANIUM-DERBY PICKLE BATH. R. S. Peoples, F. W. Fink, O. M. Stewart, and W. J. Braun. July 14, 1954. Decl. Feb. 14, 1957. 26p. Contract W-7405-eng-92. $0.35(OTS).

The inhibition of the corrosion of stainless steel in the nitric acid baths used for removal of MgF_2 from U ingots by complexing the F^- ions with metal-ion additives was studied. Al was more effective than the other metallic ions in controlling the corrosion, which may be explained by the tendency of Al to form the $[AlF_6]^{3-}$ complex. (C.W.H.)

7604 BNL-2774
Brookhaven National Lab., Upton, N. Y.
HVL-2 [REPORT. METALLURGY HIGH VELOCITY LOOP]. Metallurgy Memo 571. O. F. Kammerer. Jan. 12, 1956. Decl. Feb. 18, 1957. 10p. $0.25(OTS).

The metallurgy high-velocity loop was circulated for 1026 hours at a temperature gradient of 125°C. The maximum interface temperature at the main heater was 525°C and the minimum temperature at the exit of the finned cooler was 400°C. The flow rate was 1.2 gpm which yielded a velocity of 0.73 fps in the $\frac{3}{4}$ inch pipe and 5 fps in the right angle bend. $2\frac{1}{4}$ Chrome–1 molybdenum steel was utilized for the loop's fabrication. A G.E. G-6 electromagnetic pump was used to circulate the Bi. Gammagraphs were taken at various times during the run. No precipitation was detected. Operational data pertaining to the liquid metal treatments is summarized. (auth)

7605 CE-973
[Chicago. Univ.] Metallurgical Lab.
CORROSION OF ALUMINUM UNDER "W" PILE CONDITIONS. Progress Report on Problem 51-1A. R. B. Briggs. Oct. 9, 1943. Decl. Feb. 18, 1957. 27p. Contract [W-7401-eng-37]. (A-1380). $4.80(ph OTS); $2.70 (mf OTS).

7606 CF-51-4-20
[Oak Ridge National Lab., Tenn.]
CORROSION NEWSLETTER NO. 1. J. L. English. Apr. 4, 1951. Decl. Feb. 14, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A study was made of UO_2SO_4 solution stability at 250°C as a function of exposure time in untreated stainless steel systems. It was concluded that total U reduction from the UO_2SO_4 was nearly 100% complete in 8 hours, that no definite correlation of U^{+4} and total pH vs. total U content could be obtained, and that type 347 stainless steel showed a slight weight increase during the first four hours. The effect of Graphitar No. 14 on the stability of UO_2SO_4 at 100°C and 250°C was also studied. An average reduction of 9.9% in total U content was obtained during 24 hours at 100°C and an 18% reduction in total U during 24 hours at 250°C. (B.J.H.)

7607 CF-51-8-230
[Oak Ridge National Lab., Tenn.]
CORROSION NEWSLETTER NO. 4. J. L. English. Aug. 18, 1951. Decl. Feb. 14, 1957. 25p. $4.80(ph OTS); $2.70(mf OTS).

Successful stagnant corrosion tests on the corrosive effects of UO_2SO_4 were obtained with HNO_3-pretreated stainless steel systems for as long as six weeks of continuous operation at 250°C. Under conditions of the test, an estimate of the O available in the system to maintain passivity was 0.28 cc/cm^2 of exposed surface area. X-ray diffraction studies disclosed the stainless steel surface films to contain >80% α-Fe_2O_3 and <5% Fe_3O_4. Although the overall test results may be influenced one way or another by the initial surface condition of the "used" bombs, indications are that the oxygen requirement to maintain passivity on HNO_3 pretreated stainless steel surfaces is very slight, ≤0.3 cc/cm^2, under stagnant conditions. (auth)

7608 CF-51-9-152
Oak Ridge National Lab., Tenn.
EFFECT OF MERCURY ON THE CORROSION OF TYPE 309SCb STAINLESS STEEL, IN ACID NITRATE SOLUTIONS. F. L. Culler. Sept. 29, 1951. Decl. Feb. 20, 1957. 3p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

7609 CF-52-5-33
Oak Ridge National Lab., Tenn.
INITIAL REPORT ON THE CORROSION OF FUSED TITANIA AND SYNTHETIC SAPPHIRES IN URANYL SULFATE SOLUTIONS AT 250°C. J. L. English. May 5, 1952. Decl. Feb. 14, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); 1.80(mf OTS).

Samples of fused titania and synthetic sapphires submitted for corrosion behavior evaluation in uranyl sulfate solutions at 250°C showed weight losses of 0.2 to 0.3 mil/yr for titania and 0.7 to 0.8 mil/yr for sapphires. Of the numerous ceramics and related materials studied thus far in uranyl sulfate solutions, these have offered the most promising corrosion behavior. (F.S.)

7610 CF-52-11-76
Oak Ridge National Lab., Tenn.
CORROSION OF STAINLESS STEELS BY HNO_3–HF–$Al(NO_3)_3$ MIXTURES DURING EVAPORATION OF PLUTONIUM PRODUCT STREAMS. T. A. Arehart and T. C. Runion. Nov. 10, 1952. Decl. Feb. 14, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The rate of corrosion of stainless steels during the evaporation of Hf–HNO_3 mixtures is quite high even in the

presence of excess Al. The corrosion rate was six to ten times the rate normally considered permissible. (F.S.)

7611 CF-53-7-100
Oak Ridge National Lab., Tenn.
THE EFFECT OF RUN DURATION, SAMPLING FREQUENCY, AND NICKEL PICKUP ON THE UNCERTAINTY IN OBSERVED HRE CORROSION RATES. E. L. Compere. July 16, 1953. Decl. Feb. 14, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Revisions in the proposed HRE corrosion experiments are briefly discussed. (B.J.H.)

7612 CF-54-6-7
Oak Ridge National Lab., Tenn.
CORROSION OF TYPES 347 AND 309 SCb STAINLESS STEELS BY SIMULATED "25" PROCESS WASTE SOLUTIONS AT VARIOUS TEMPERATURES. Arnold R. Olsen. June 1, 1954. Decl. Feb. 14, 1957. 40p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

Corrosion tests on stainless steels Type 347 and Type 309 SCb exposed to concentrated simulated first cycle raffinate (predominantly 2.5 molar aluminum nitrate) from "25" Process at Arco at temperatures from 120°F to boiling have been made. The corrosion rates are distinctly temperature dependent. Severe intergranular attack occurred at temperatures above 180°F disappearing at 150°F for Type 347 stainless steel and at 120°F for Type 309 SCb stainless steel. From the results of these tests it appears that successful long time storage of this waste will require that the solution temperature be maintained below 150°F. (auth)

7613 CT-1182
[Chicago, Univ.] Metallurgical Lab.
CORROSION. PART IV OF REPORT FOR MONTH ENDING JANUARY 1, 1944. Decl. Feb. 18, 1957. 18p. Contract [W-7401-eng-37]. (A-1663). $4.80(ph OTS); $2.70(mf OTS).

High weight losses are shown by 2S Al at pH's less than 4 to 4.5 and pitting below 5 to 5.5 at 1×10^{-4} N H_2O_2 and 1 ppm Cl^- in simulated W water. 72S Al corrodes about like 2S, with perhaps a bit more attendant pitting. Statistical tests showed no pitting at pH 7.7 to 8.2, with penetration rate up to about 0.3 mil/mo. Seam-welded caps pitted through under the crimp at 90°C, 1×10^{-4} N H_2O_2 and 2 ppm Cl^-, but showed no signs of attack in the absence of Cl^-. Arc-welded caps show no attack except corrosion around inclusions which probably are emery. 18-8 stainless steel causes 2S Al to pit severely when coupled to it. No evidence of 2S strain corrosion could be detected in a static system, pH = ~6. (auth)

7614 HW-17010
Hanford Works, Richland, Wash.
CORROSION OF AUSTENITIC STAINLESS STEEL TYPE 347 IN REDOX PROCESS STREAMS. (ORNL #1 FLOWSHEET). W. W. Koenig. Feb. 23, 1950. Decl. Feb. 19, 1957. 10p. Contract W-31-109-eng-52. $1.80(ph OTS); $1.80(mf OTS).

Austenitic stainless steel, type 347, was exposed in various welded samples which had been stressed, pickled, annealed, sandblasted, etc. to ORNL No. 1 Redox streams IAF, IAS, and IDS. Under these dynamic and alternate-immersion laboratory corrosion test conditions at room temperature, this stainless steel was found to exhibit excellent corrosion resistance. "As welded" specimens which had been prepared by welding T-347 stainless steel to T-309 stainless steel showed the same excellent corrosion resistance to these Redox streams in laboratory alternate-immersion corrosion tests at room temperature. (auth)

7615 HW-17626
Hanford Works, Richland, Wash.
CORROSION OF AUSTENITIC STAINLESS STEELS IN MJ-4 RAW STREAMS. W. W. Koenig. April 25, 1950. Decl. Feb. 19, 1957. 14p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

The corrosion resistance of several austenitic stainless steels to the RAW stream of the Tributyl Phosphate Waste Metal Recovery Process was studied. Carpenter 20 and T-309 SCb steels showed good resistance, while T-304 ELC showed good to fair resistance, and T-347 showed fair resistance. (B.J.H.)

7616 HW-30500
General Electric Co. Hanford Atomic Products Operation, Richland, Wash.
THE RESISTANCE OF VARIOUS BEARING MATERIALS TO CHEMICAL ATTACK BY NITRIC ACID AND URANYL NITRATE HEXAHYDRATE. Norman D. Groves. Mar. 11, 1954. Decl. Feb. 19, 1957. 20p. Contract [W-31-109-eng-52]. $3.30(ph OTS); $2.40(mf OTS).

The chemical resistance of various bearing materials, used in Purex submerged pumps, to solutions of HNO_3 and $UO_2(NO_3)_2$ was studied. Results indicated that Corning glass, boron carbide, titanium, or Chemelac M1-411 exhibited sufficient resistance to chemical attack by 60% HNO_3 and by 100% uranyl nitrate solutions at boiling temperatures to warrant their use as bearings in Purex submerged rotating equipment. (J.E.D.)

7617 HW-37049
Hanford Atomic Products Operation, Richland, Wash.
SURFACE STRUCTURE OF PILE IRRADIATED 2S ALUMINUM. R. Borasky and B. Mastel. Aug. 1, 1955. Decl. Feb. 7, 1957. 24p. Contract W-31-109-eng-52. $4.80(ph OTS); $2.70(mf OTS).

The surfaces of cold rolled, half hard, nitric acid treated, autoclave etched, 2S Al coupons were studied by optical and electronoscopic methods, before and after exposure to hot water alone and hot water and neutron bombardment. The non-exposed aluminum surfaces have a characteristic etch pattern which is absent in the exposed coupons. All of the exposed coupons were corroded. In general the coupons exposed to the higher flux intensities of irradiation were more severely corroded than those exposed to the lower flux intensities of irradiation or to the hot water alone. Representative photo- and electron micrographs illustrate the various types of surface structure observed on the control and experimental coupons. (auth)

7618 MonT-348
Clinton Labs., Oak Ridge, Tenn.
CORROSION OF 2S ALUMINUM IN DEMINERALIZED WATER. Progress Report on Problem Assignment No. TX5-12. D. G. Reid and L. P. Bornwasser. July 24, 1947. Decl. Feb. 26, 1957. 13p. Contract W-35-058-eng-71. $3.30(ph OTS); $2.40(mf OTS).

Preliminary corrosion tests have been completed with 2S Al exposed to demineralized H_2O at H_2O velocities from 0 to 40 ft/sec, H_2O_2 concentrations from 0.05 to 0.0005 molar with the pH maintained at 6, and pH from 3.0 to 6.5 with the peroxide concentration maintained at 0.005 M. The corrosion was studied at temperatures between 80 and 110°C. Corrosion rates as determined by weight losses were generally low. The highest rate obtained in these tests was 10^{-3} in/mo. The pitting was moderate to severe; the maximum pitting rate observed was 0.018 in. in 166 hrs. (auth)

7619 ORNL-942
Oak Ridge National Lab.; Tenn.
THE CORROSION OF HOT-PRESSED, EXTRUDED, AND

VACUUM CAST BERYLLIUM IN SIMULATED COOLING WATER FOR THE MATERIALS TESTING REACTOR. James Reed. Jan. 9, 1953. Decl. Mar. 2, 1957. 90p. Contract W-7405-eng-26. $0.60(OTS).

The results of stagnant tests to determine the effects of production method, surface cold work, water quality, and metal purity on the corrosion resistance of Be to aqueous media are reported. Distilled water, with and without added ionic impurities, containing H_2O_2 (0.005M) at 85°C was the corrosive medium. (F.S.)

7620 ORNL-947
Oak Ridge National Lab., Tenn.
THE EFFECT OF FLUORIDE ON THE CORROSION OF STAINLESS STEEL IN THE PRESENCE OF EXCESS AL^{+3} ION. Arnold R. Olsen. Mar. 27, 1951. Decl. Feb. 7, 1957. 9p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

Corrosion tests of Type 347 stainless steel for two weeks in boiling uranyl nitrate, sulfuric acid media, with and without additions of Al^{+3} and F^{-1} in a 2:1 mole ratio, were made. The results show that although the presence of aluminum greatly reduces the deleterious effect of fluoride, the corrosion rates are still three to four times greater than when no fluoride is present. (auth)

7621 ORNL-1066
Oak Ridge National Lab., Tenn.
THORIUM AND THORIUM ALLOYS PRELIMINARY CORROSION TESTS. Arnold R. Olsen. Dec. 19, 1951. Decl. Mar. 4, 1957. 21p. Contract W-7405-eng-26. $4.80 (ph OTS); $2.70(mf OTS).

The corrosion resistance of thorium and thorium alloys containing from 2 to 6% chromium, niobium, titanium, zirconium and titanium–zirconium in distilled water at 95°C was investigated. The true effects of the various alloy additions could not be determined because of the more marked effect of cold working. It was found that a sample cold rolled 75% and annealed would show a tenfold higher weight change over an otherwise identical sample cold rolled 68% and annealed; or a decided loss in weight compared to a relatively small gain. (auth)

METALS AND METALLURGY

7622 ANL-4450
Argonne National Lab., Lemont, Ill.
CORROSION TEST PROCEDURE FOR THE SELECTION OF CRYSTAL BAR ZIRCONIUM FOR USE IN NAVAL REACTOR FUEL ELEMENTS. (Includes Addendum). V. F. Saitta. May 10, 1950. Decl. Feb. 11, 1957. 12p. Contract W-31-109-eng-38. $3.30(ph OTS); $2.40(mf OTS).

Reliable test equipment and a reproducible corrosion test procedure have been developed for the inspection and classification of crystal bar Zr for naval reactor fuel elements. Every bar is examined and classified by a 168-hour high-temperature water corrosion test. This procedure has been instituted by ANL and is in use by the producers of crystal bar Zr, the Foote Mineral Company. (auth)

7623 ANL-5076
Argonne National Lab., Lemont, Ill.
DEVELOPMENT OF ZIRCONIUM CLAD URANIUM PLATES FOR REACTOR FUEL. J. J. Lawless, C. H. Bean, J. R. Wooland, and R. E. Macherey. May 6, 1953. Decl. Feb. 14, 1957. 76p. Contract W-31-109-eng-38. $0.55(OTS).

A method was developed for production of Zr-clad U plate based upon previous investigations at Argonne and Oak Ridge National Laboratories, and preliminary investigations in the present work. The present preliminary investigations included experimental studies of the effect of interface atmosphere upon roll bonding and led to the conclusion that thin layers of air between bonding surfaces could be tolerated. It was also determined experimentally that rectangular billets rolled in flat rolls were most promising for production of plates with good dimensional characteristics. A method of billet manufacture was devised in which the core and jacket parts were assembled, pressed together to minimize the amount of air at the core-jacket interface, and all jacket seams welded while under pressure. The billet was then sheathed in a steel can for protection of the Zr jacket during hot rolling. Several canning steels were tested, and it was found that steel containing a gas-stabilizing alloy such as Ti had the least affect on the ductility of the cladding. Various rolling temperatures were studied, and different combinations may be recommended depending upon the properties desired in the finished plate. Results of a considerable amount of testing show that the original microstructure of the core material will have a significant effect on cladding unformity and point up the requirement for well designed rolling equipment and skillful operation if plates with close dimensional tolerance are to be produced. (auth)

7624 ANL-5117
Argonne National Lab., Lemont, Ill.
PREFERRED ORIENTATION IN 300°C ROLLED AND IN RECRYSTALLIZED URANIUM SHEET. Melvin H. Mueller, Harold W. Knott, and Paul A. Beck. Sept. 15, 1953. Decl. Feb. 7, 1957. 45p. Contract W-31-109-eng-38. $7.80(ph OTS); $3.30(mf OTS).

The rolling and recrystallization textures in 300°C rolled uranium sheet were investigated using a Geiger counter diffractometer with the modified Schulz reflection technique. Seven sections of sheet material were used in order to obtain sufficient data for quantitative pole figures by the reflection technique. A special integrating specimen table was used for obtaining and recording the data automatically. The rolling texture was described in terms of several "ideal orientations" in which the stronger orientations have either the (110) or (010) poles in the vicinity of the rolling direction. The recrystallization texture of the sheet uranium was also described by several "ideal orientations" which are somewhat different, except in one case, than the as rolled "ideal orientations." Either approximate (010), (140), or (553) poles are near the rolling direction in the recrystallized texture. (auth)

7625 ANL-5180
Argonne National Lab., Lemont, Ill.
ROLL CLADDING OF URANIUM–NIOBIUM ALLOYS FOR PLATE TYPE FUEL ELEMENTS. C. H. Bean and R. E. Macherey. Dec. 1953. Decl. Feb. 12, 1957. 59p. Contract W-31-109-Eng-38. $0.50(OTS).

The feasibility of cladding plate type U–Nb fuel elements with Zircaloy-2 by roll bonding has been demonstrated. Plates with cores of U alloyed with 3 and 6 wt. % Nb have been finished within specified tolerances. Plates can be made with clad to core bond strengths from 30,000 to 60,000 psi. Properly heat-treated plates have sufficient ductility to allow cold finishing by rolling, forming, bending, or twisting. Fabrication procedures are described. Clad plates were inspected by metallography, autoradiography, physical measurements, mechanical testing, and corrosion testing, and results are summarized. (M.P.G.)

7626 ANL-5234
Argonne National Lab., Lemont, Ill.
METALLURGY DIVISION QUARTERLY REPORT [FOR] OCTOBER, NOVEMBER, AND DECEMBER 1953. Dec. 31, 1953. Decl. Feb. 12, 1957. 78p. Contract W-31-109-Eng-38. $12.30(ph OTS); $4.50(mf OTS).

Progress is reported on the following studies: cold drawing of extruded and pierced Zr tubing; resistance welding of roll-formed Zr; cladding of U–Nb alloys with Zircaloy II; resistance roll cladding of U with Zr; fabrication, heat treatment, and thermal cycling of EBR fuel blanket slugs (U–Zr alloy); fabrication of U fuel plates for fast exponential experiment; melting and casting of U–Zr alloys from co-reduced Mallinckrodt biscuit; design of a high-temperature tensile apparatus and tensile strength of Zr- and Ti-base alloys (Ti–Al–Cr, Ti–Al–Mn, Ti–Cr–Fe, Ti–Mn, Zr–Al, and Zr–Ti alloys); transformation and melting point temperatures of high-purity U by thermal analysis; twinning and slipping in compressed U single crystals; preferred orientation of U rods rolled at 300 to 640°C, high-purity U rods rolled at 300°C and recrystallized, and a β-treated U sheet at 740°C and water quenched; phase studies and microstructure of C–U alloys; solubility of Zr in α uranium and microstructures of U–Zr alloys quenched from 800 to 1000°C; microstructure and hardness of U–Zr–Nb alloys; zone melting of U alloys; diffusion coating of U with Si; effects of irradiation on the dimensions of natural and α uranium single crystals polycrystalline natural U, and enriched U–Zr alloys; and effects of irradiation on 347 stainless steel-clad U in the MTR reflector stringer. (J.A.G.)

7627 ANL-5296

Argonne National Lab., Lemont, Ill.

MECHANICAL PROPERTIES OF ROLLED URANIUM RODS. Final Report–Metallurgy Program 4.1.9. Work completed: September 1953. R. M. Mayfield, H. H. Chiswik, and R. E. Macherey. July 1, 1956. Decl. Feb. 14, 1957. 42p. Contract W-31-109-eng-38. $0.40(OTS).

The mechanical properties of alpha U rolled rods were examined. It was found that mechanical strength varies with the temperature of rolling. The ultimate strength and yield strength values decrease linearly with increasing rolling temperature. The ductility as measured by the elongation in the rolling direction increases linearly with rolling temperature. The highest ductilities are obtained by rolling at temperatures above the recrystallization temperature. A series of tension and compression tests have been performed on normal U test bars rolled and annealed in the α temperature and β treated. An experiment has been performed to determine the degree of grain growth or coarsening and the danger of transforming into β phase on soak heating as-rolled U rods. (W.L.H.)

7628 ANL-5377

Argonne National Lab., Lemont, Ill.

MECHANICAL STABILIZATION OF URANIUM FUEL ELEMENTS. Final Report—Metallurgy Program 5.1.12. K. F. Smith, S. T. Zegler, and R. M. Mayfield. Nov. 1956. Decl. Feb. 11, 1957. 35p. Contract W-31-109-Eng-38. $0.40(OTS).

A study of the mechanical restraint of cylindrical specimens of U under thermal cycling was concluded. Various degrees of restraint were accomplished by the use of containing jackets of different materials, with occasional assistance of minor alloying of the U with Zr. The ratio of clad cross-sectional area to core cross-sectional area is a means employed for expressing the effectiveness of restraint provided by a given material. Unalloyed Zr was found ineffective as a jacketing material in any thickness for use at 550°C or higher. Type 347 stainless steel was an effective restraining material to 800°C in clad-to-core area ratios of 22% or higher, provided means are used to provide mechanical separation of clad and core above 730°C to prevent alloying between U and steel. Tantalum is an effective restraining material to 800°C in clad-to-core area ratios as low as 11%, and probably as low at 7%. Although not suitable for reasons of physics for thermal type reactors, results suggest investigation of Nb, which has similar properties, for those applications. Considerably reduced propensity for growth, increased grain size, and increased thermal expansion have been observed for U which has been thermal cycled under complete restraint. The findings on minimum acceptable jacket thickness for each material are summarized. (auth)

7629 ANL-5489

Argonne National Lab., Lemont, Ill.

METALLURGY DIVISION QUARTERLY REPORT [FOR] JULY, AUGUST, AND SEPTEMBER 1955. Mar. 1956. Decl. Feb. 12, 1957. 67p. Contract W-31-109-eng-38. $0.50(OTS).

Progress is reported on the following investigations: fabrication and evaluation of uranium alloy fuel plates for EBWR; radiation damage in EBR-II slugs; radiation effects on U–Pu and U–Si alloys, U powder compacts, U, and UO_2; corrosion resistance of Nb–U–Zr, U–Si, Al–Ni, and Al alloys; preparation of single crystals of alpha U; phase studies and microstructures of U–Zr and U–Nb–Zr alloys; and vacuum fusion determination of H_2 in Al. (For preceding period see ANL-5465.) (C.W.H.)

7630 ANL-5541

Argonne National Lab., Lemont, Ill.

METALLURGY DIVISION QUARTERLY REPORT [FOR] OCTOBER, NOVEMBER, AND DECEMBER 1955. June 1956. Decl. Feb. 12, 1957. 55p. Contract W-31-109-eng-38. $0.50(OTS).

The fabrication and evaluation of U alloy fuel plates for EBWR, properties of Mo–Pu–U alloys, the development of fuel elements and structural, shielding and control rod materials, preparation of U single crystals for elastic constants studies, Nb–U–Zr alloys phase diagram, solubility of C in gamma U, thermal conductivity apparatus for U, corrosion mechanism of Al at elevated temperatures, neutron-diffraction study of Ti–Zr alloys, and the neutron flux distribution and Wilkins effect in U rods are reported. (For preceding period see ANL-5489.) (J.E.D.)

7631 BBC-7

Brush Beryllium Co., Cleveland.

PROGRESS REPORT [FOR] FEBRUARY 1 TO FEBRUARY 29, 1948. D. T. Doll. Decl. Feb. 16, 1957. 30p. Contract AT-30-1-Gen-155. $4.80(ph OTS); $2.70(mf OTS).

Fin Tubes. One run was made in which an 8 fin tube, 24 inches long, was sintered from Be metal powder. The method wherein gas pressure was applied externally to the metal envelope during sintering was used. The sintered tube was tested in water under 100 psi He pressure. A length of approximately 12 inches was pressure tight. The balance of the tube showed porosity. Rods. In Progress Report BBC-2 the production of rods from sintered Be metal was discussed. This work had utilized a split steel supporting mold to hold the metal tubes containing the Be powder. The present work used a graphite mold in place of the steel mold. The graphite fractured during the run. Failure of the graphite mold was thought to be due to differences between the steel tubes containing the powder and the graphite. The sintered rods had a density of 1.86. Two of the rods had surface imperfections due to loss of metal powder during evacuation. The faulty portion of the machined rod was discarded. The other four rods were machined to size and were quite satisfactory. G. E. Washers. Five runs were made to study methods for the production of G. E. washers. Three of the runs were made to examine the suitability of Mo as a plunger material to hot press washers. After the third hot press run, the Mo plungers showed that alloying had occurred between the plungers and the Be metal. Density of the sintered washers produced in these

runs varied. The washers showed good surface detail, however, the detail was not as sharp in the third washer as was shown in the first washer. In two cold press and sinter runs, the powder was compacted under 60 tons per $in.^2$ pressure in the first run and under 80 tons per $in.^2$ in the second run. The green compacts were placed in cavities in a steel plate. A second steel plate was used to cover the holding plate. The sintering tube was evacuated and the washers sintered at 1040°C. for 12 hours. The sintered washers from both runs were coated with a thin oxide scale. Measurement of the oxide scale thickness of washers from the second run showed the scale to be 0.0004 inch thick. (auth)

7632 BMI-76

Battelle Memorial Inst., Columbus, Ohio.

THE TECHNOLOGY OF THORIUM. L. L. March, Jr. and J. R. Keeler. July 18, 1951. Decl. Feb. 14, 1957. 32p. Contract W-7405-eng-92. $0.45(OTS).

A critical survey is presented of information available at the present time in the project literature and unpublished data on production methods, fabrication techniques, and chemical, physical, and metallurgical properties of Th metal. 56 references. (C.H.)

7633 BMI-868

Battelle Memorial Inst., Columbus, Ohio.

TERNARIES OF THE EUTECTIC URANIUM–CHROMIUM ALLOY. H. A. Saller, F. A. Rough, and A. A. Bauer. Sept. 21, 1953. Decl. Feb. 20, 1957. 18p. Contract W-7405-eng-92. $0.25(OTS).

A study has been made of the effect of small additions of Fe, Ni, and Mn to the U–5 wt. % Cr eutectic alloy. The effect upon melting point, allotropic-transformation temperatures, ductility, and general usefulness has been investigated. A ternary eutectic is produced upon addition of 1 wt. % Fe to the base alloy with a reduction in melting point of 108°C. The addition of Ni or Mn causes a gradual decrease in the liquidus of the alloy, 2 wt. % causing decreases of 40 and 58°C, respectively, while the solidus temperatures are lowered 124 and 100°C, respectively. Both Fe and Ni produce a gradual lowering of the α-β and β-γ transition temperatures of the binary Cr eutectic alloy. These are lowered 6 and 5°C, respectively, upon addition of 1 wt. % Fe, and 15 and 19°C, respectively, upon addition of 1 wt. % Ni. The addition of 0.5 wt. % Mn lowers the transition temperatures 26 and 21°C, respectively. A reversal of effect then occurs as further Mn additions cause a gradual increase in these temperatures. The ductility and workability of these alloys decrease with increasing ternary additions. The Fe and Ni ternaries, however have greater ductility than do the corresponding Mn ternaries. (auth)

7634 BMI-884

Battelle Memorial Inst., Columbus, Ohio.

PREPARATION AND PROPERTIES OF THE EUTECTIC URANIUM–CHROMIUM ALLOY. H. A. Saller, F. A. Rough and R. F. Dickerson. Nov. 3, 1953. Decl. Feb. 14, 1957. 22p. Contract W-7405-eng-92. $0.30(OTS).

The process of casting eutectic U–Cr ingots and the resulting microstructures and densities have been studied. Data on shrinkage during solidification, fabricability, mechanical properties, and specific heat of this alloy were also obtained. In addition, the constitutional diagram of the U–Cr system has been briefly reviewed. The eutectic U–Cr alloy (about 5.0 wt.%) was found to be cast easily under a variety of casting conditions. Measured shrinkage during solidification was in the neighborhood of 4.5 to 5.0%. The density of the cast alloy was found to be close to theoretical density. The structures of the eutectic alloy were affected little by casting variables, but were altered by the presence of primary Cr. Specimens of the material were found to be readily fabricable by rolling at 1050°F. Tensile strengths of the eutectic alloy in the as-cast condition were found to be as high as 121,900 psi. The dynamic modulus of elasticity was 29.2×10^6 psi at 21°C and decreased with increasing temperature. The mean specific heat of a 5.6 wt.% Cr sample was found to increase linearly with an increasing upper limit of the temperature range up to 625°C. (auth)

7635 BMI-992

Battelle Memorial Inst., Columbus, Ohio.

ELECTRODEPOSITION OF ALUMINUM ON URANIUM. J. G. Beach, W. C. Schickner, Delores Hopkinson, and C. L. Faust. Mar. 30, 1955. Decl. Feb. 7, 1957. 18p. Contract W-7405-eng-92. $0.25(OTS).

Aluminum electroplating was studied in a search for new methods of cladding U fuel elements. Uranium electroclad with 12 mils of Al over a 0.5-mil Ni (or Ni plus Cu) electroplate resisted corrosion for more than 100 hr in boiling water. This quality of protection was effected by hot pressing the electroclad U with 5.1 tons/$in.^2$ for 5 min at 950°F. The electroclad U without the hot pressing generally failed within 24 hr. The results with hot-pressed samples paralleled those of later experiments with hot-pressed wrought Al claddings on U. In both cases, the U was electroplated with thin (0.5 mil) layers of metals to prevent Al–U diffusion, to aid bonding, and to assist in corrosion protection. This Al electroplating study helped to define the importance of the intermediate coating between the Al and the U, the effect of good bonds between the various layers, and the effects of hot pressing in protecting U with an Al cladding. (auth)

7636 BMI-999

Battelle Memorial Inst., Columbus, Ohio.

GRAIN REFINEMENT OF THE AS-CAST URANIUM - 5 w/o CHROMIUM ALLOY BY HEAT TREATMENT. Henry A. Saller, Frank A. Rough, and Arthur A. Bauer. May 13, 1955. Decl. Feb. 13, 1957. 14p. Contract W-7405-eng-92. $0.25(OTS).

The effect of heat treatment on the as-cast grain size of the uranium-5 w/o chromium alloy has been studied. Gamma or β treatments followed by isothermal transformation at α temperatures are effective in refining grain size. The most effective treatment which was tried consists of annealing 1 min at 780°C followed by isothermal transformation for 2 hr at 475°C. A combination of heat treatment with chill casting of the alloy may be expected to effect additional refinement. (auth)

7637 BMI-1001

Battelle Memorial Inst., Columbus, Ohio.

THERMAL CYCLING OF SPECIMENS OF URANIUM-CHROMIUM ALLOYS. Henry A. Saller, Ronald F. Dickerson, and William E. Murr. May 18, 1955. Decl. Feb. 13, 1957. 30p. Contract W-7405-eng-92. $0.35 (OTS).

The effects of thermal cycling upon specimens of the uranium-chromium eutectic alloy (5 wt.% chromium) were studied. Cylinders of the alloy were thermally cycled over three temperature ranges, 300 to 1022°F, 300 to 1275°F, and 300 to 1450°F. Included are studies of both hot and cold cast specimens. In addition, effects of thermal cycling upon specimens of noneutectic compositions of uranium-4.5 wt.% chromium and uranium-5.5 wt.% chromium were studied. Generally, the noneutectic alloys withstood cycling deformation better than specimens of the eutectic composition. Specimens that were cold cast had smoother appearances after being cycled from 300 to 1022°F than specimens that were cast in heated molds. (auth)

7638 BMI-1028

Battelle Memorial Inst., Columbus, Ohio.

THE KINETICS OF THE ZIRCONIUM–URANIUM DIOXIDE REACTION. Manley W. Mallett, Arnold F. Gerds, Alexis W. Lemmon, and Dan L. Chase. Aug. 15, 1955. Decl. Feb. 18, 1957. 70p. Contract W-7405-eng-92. $0.50(OTS).

The kinetics of the zirconium–UO_2 reaction was studied using crystal-bar zirconium and fused UO_2 high-fired in hydrogen. The reaction was studied with solid zirconium cylinders and UO_2 powder, compacted mixtures of zirconium and UO_2 powders, and sandwich-type elements. Experiments performed with sandwich-type elements yielded the most useful data both from a practical and a fundamental point of view. Exploratory experiments designed to give a wider range of experimental materials were unsuccessful in producing substoichiometric UO_2. Results obtained from sandwich-type experiments showed that slow reactions at 950 and 1100°F formed reaction zones less than 0.0002 in. thick in 65 days. A measurable reaction took place at 1300°F, while at 2000°F, the reaction was very rapid. The UO_2 seems to be reduced to its elements, uranium and oxygen, which diffuse simultaneously, but not with equal speed, into the zirconium. Reactants heated at 1300°F and slowly cooled to room temperature, were found to have formed alpha zirconium and alpha uranium adjacent to the UO_2. Deeper in the original zirconium region the epsilon uranium–zirconium intermetallic phase was found dispersed in a zirconium–oxygen solid solution. Elements slowly cooled from 2000°F contained alpha zirconium and alpha uranium at the UO_2–metal interface. A continuous stained band having the alpha uranium lattice structure and containing some zirconium was observed beneath the two-phase region. Oxygen had diffused beyond the uranium-containing region, forming the hard zirconium–oxygen solid solution. On the basis of limited data obtained, it has been estimated that penetration of oxygen through the 0.030-in.-thick zirconium jacket walls probably would cause their destructive embrittlement in about 300 days at 1300°F or 2½ days at 2000°F. The reaction between Zircaloy 2 and UO_2 at 1300°F appears to be somewhat less rapid than that between unalloyed zirconium and UO_2. The phases formed by this reaction have not been identified. (auth)

7639 BMI-1030

Battelle Memorial Inst., Columbus, Ohio.

AN EVALUATION OF DATA ON ZIRCONIUM–URANIUM ALLOYS. Frank A. Rough. Aug. 19, 1955. Decl. Feb. 14, 1957. 112p. Contract W-7405-eng-92. $0.70(OTS).

A survey of the literature relative to Zr–U alloys was made. The available data were analyzed in an attempt to present the most coherent and reliable picture of the behavior and nature of the Zr–U alloys as a system. The constitutional diagram was reviewed, and an attempt was made to reconcile the divergent forms of this diagram found in the literature. The structure of the intermediate phase, epsilon, is discussed. The two modes of transformation of the beta (high temperature) phase are described; one mode resembles the martensite transformation found in steels; the other mode resembles in some respects the age-hardening-type transformation found in Ti alloys. Typical microstructures produced by transformation of beta, by cold working and annealing, and by duplex heat treatments are shown. An attempt was made to correlate structure with the manner in which it was produced and with the resultant mechanical properties. The available data on the hardness, tensile, and creep properties of Zr–U alloys are tabulated and evaluated in the light of recent developments in the field of transformation kinetics. The physical properties of the alloys are presented in graphical form. The best available techniques for hot rolling, extruding, cold rolling, and swaging are detailed. The corrosion resistance of Zr–U alloys in water at various temperatures is reviewed. The effect of ternary additions upon corrosion resistance is discussed with a view toward improving the alloys by this means. The effect of neutron irradiation upon metals is discussed, in general with respect to fuel alloys, and specifically with respect to Zr–U alloys. (auth)

7640 BMI-1048

Battelle Memorial Inst., Columbus, Ohio.

THE CHARACTERISTICS OF THE BOND INTERFACE FORMED BETWEEN ZIRCALOY 2 AND URANIUM-12 w/o MOLYBDENUM. Henry A. Saller, Stan J. Paprocki, and Edwin S. Hodge. Oct. 17, 1955. Decl. Feb. 26, 1957. 50p. Contract W-7405-eng-92, $0.45(OTS).

The interface formed between a uranium–12 w/o molybdenum core alloy and Zircaloy 2 was investigated to determine its structure and corrosion behavior in high-temperature water. Assembled picture-frame-type elements were pressure bonded under conditions which yielded various degrees of interdiffusion. The bonded specimens were defected through the Zircaloy 2 cladding into the alloy core with a 40-mil hole and corrosion tested in 650°F water. Those bonds which possessed a minimum amount of diffusion were most corrosion resistant. The corrosion resistance decreased progressively with increased interdiffusion. Specimens with extensive bond diffusion failed by rupturing at the interface in less than 1 hr of exposure in 650°F water. The various alloy layers of the interface were identified and individual alloys were prepared and evaluated in high-temperature water. The alloy layers which exhibited poor corrosion resistance in the interface also possessed poor resistance when tested as discrete alloys. The least-resistant alloys were those possessing 12 w/o molybdenum with small amounts of zirconium replacing the uranium of the base core alloy. Transition layers between the uranium–12 w/o molybdenum core alloy and Zircaloy 2 were investigated in an effort to obtain a more resistant interface than was obtained by direct bonding. Of the layers that were investigated, chromium, platinum, and tantalum were the best, but none of these layers was sufficiently effective to warrant further study. An exploratory study to determine the effect of different barrier layers on the interdiffusion of Zircaloy 2 and uranium–3.8 w/o silicon alloy was also made. (auth)

7641 BMI-1053

Battelle Memorial Inst., Columbus, Ohio.

CENTRIFUGAL CASTING OF PLATE-TYPE FUEL ELEMENTS. Henry A. Saller, Ronald F. Dickerson, and William E. Murr. Nov. 10, 1955. Decl. Feb. 13, 1957. 18p. Contract W-7405-Eng-92. $0.30(OTS).

A study was made on production of plate-type elements by centrifugal casting methods. Alloys investigated were uranium–5 wt.% chromium, uranium–5 wt.% chromium–1 wt.% molybdenum, and uranium–2 wt.% zirconium. Grain refinement of the eutectic alloy (uranium–5 wt.% chromium) was accomplished by casting into metal molds. However, the largest section of sound casting obtained was 5½ in. long. The uranium–2 wt.% zirconium alloy produced unsatisfactory castings because of its poor fluidity. This resulted in castings with numerous cold shuts and seams and poor detail. On the basis of 14 castings, it does not appear feasible to produce plate-type fuel elements by the methods employed. Refractory mold and downsprue components would largely eliminate problems encountered with metal molds. For this reason, the probability of producing sound centrifugally cast plates of the required size and composition by utilizing refractory components is good. (auth)

7642 BMI-1084
Battelle Memorial Inst., Columbus, Ohio.
THE EFFECTS OF CHEMICAL IMPURITIES ON THE QUALITY OF ROLLED URANIUM ROD. Henry A. Saller, John R. Keeler, and Lee J. Cuddy. Apr. 18, 1956. Decl. Feb. 12, 1957. 19p. Contract W-7405-eng-92. $0.20 (OTS).

The effects of N_2, slag, and H_2 additions on the surface quality of rolled U rod were studied. The addition of 400 ppm of nitrogen resulted in severe striations. Slag inclusions elongated during rolling and produced short open seams in the etched surface. The effects of 9 to 10 ppm of H_2 were obscured by defects in the original castings. These defects were caused by oxide films entrapped in the ingot. (auth)

7643 BMI-X-128
Battelle Memorial Inst., Columbus, Ohio.
MINUTES OF THE MEETING ON THE ANALYSIS AND METALLOGRAPHY OF HYDROGEN IN URANIUM, HELD JUNE 22, 1955. M. W. Mallett and C. B. Griffith, eds. Sept. 15, 1955. Decl. Feb. 14, 1957. 76p. $0.55(OTS).

The minutes of the meeting on the analysis of H in U are reported covering description of samples and a summary of analytical data in tables and graph form, individual laboratory reports on analytical techniques, and metallographic reports by individual laboratories. (J.E.D.)

7644 BNL-2773
Brookhaven National Lab., Upton, N. Y.
HVL-1 REPORT. [METALLURGY HIGH VELOCITY LOOP]. Metallurgy Memo 562. R. P. Risch. Sept. 26, 1955. Decl. Feb. 18, 1957. 10p. $0.25(OTS).

Operational data pertaining to the liquid metal treatments and a list of the results of the liquid metal samples taken during the run of the high-velocity loop are summarized. (J.E.D.)

7645 BRB-8
Bridgeport Brass Co., Bridgeport, Conn.
URANIUM HOT HAMMER TESTS FOR EXTRUSION. M. T. McGowan and R. M. Treco. Oct. 14, 1955. Decl. Feb. 20, 1957. 17p. Contract AT(30-1)-1405. $0.30(OTS).

Extrusion of α phase uranium appears possible between 600 and 650°C. At 650°C. similar extrudability values for U and 85:15 brass are exhibited. Extrusion of U in the beta range (660 to 760°C.) is considered difficult because of the high resistance to deformation. Alpha phase reversion from the β by quenching decreases this resistance and appears to account for the observed deformation in β range testing. Uranium at temperatures in the γ range is highly ductile and soft. (auth)

7646 BRB-10
Bridgeport Brass Co., Conn.
MONTHLY PROGRESS REPORT [FOR] JUNE 1955. July 15, 1955. Decl. Feb. 20, 1957. 20p. Contract AT(30-1)-1405. $0.30(OTS).

The results of a meeting on extruded hollow slugs are summarized. Yields from the last production extrusion were higher than previous lots. A number of uranium tubes 2.25" O. D. × 180" wall were extruded for Savannah River. About two hundred thorium rods were extruded. Interim reports on uranium rod drawing have been completed covering drawing procedures and metal properties. An extrusion of Zircaloy process tubes for HAPO showed great improvement in the quality of the ribs. A report on annealing effects of the current Zircaloy stock was completed. Drawing of shells coated with phosphate and a new proprietary lube showed very promising results with few failures. Investigation of improved lubricants for Zircaloy is continuing with evaluation on deep drawn cans. (auth)

7647 CF-50-6-185
Oak Ridge National Lab., Tenn.
METALLURGICAL PROCESSES–SF SUPPLEMENTAL MANUAL–Procedure No. 7. June 7, 1950. Decl. Feb. 20, 1957. 3p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

From time to time the facilities of the Laboratory are utilized for the production of various alloys of uranium for use throughout the project. The operations performed include melting, alloying, casting, extrusion, and machining. For the purpose of illustration a typical project, the manufacture of 2% U–Al alloy fuel rods for the Argonne CP-3 reactor, is described below. It is similar in most respects to other metallurgical procedures undertaken at Oak Ridge National Laboratory. (auth)

7648 CF-50-12-23
[Oak Ridge National Lab., Tenn.]
METALLURGICAL FACTORS AFFECTING THE DISSOLVING RATE OF 7.5% U–92.5% Al ALLOY. D. E. Hamby. Dec. 6, 1950. Decl. Feb. 20, 1957. 11p. Contract [W-7405-eng-26]. $0.25(OTS).

It was shown that there is a marked difference in the dissolving rate of cast slugs and extruded slugs of U–Al alloys. A metallographic study of the slugs reveals a possible explanation of the observed results. The compound UAl_4 is known to be dissolved by acid more rapidly than is pure Al. The UAl_4 compound particles are distributed differently depending upon the mode of solidification and fabrication. The rate of solution may depend to a great extent upon the relative surface areas of the Al and UAl_4. Photomicrographs show the distribution of the eutectic in the cast structure and in the extruded material. In the extruded material the compound particles are more or less uniformly distributed in a matrix of Al, whereas in the cast slug, the compound is present in the form of a eutectic network around the grain boundaries of pure Al. The acid can attack the alloy as a whole more readily if the easily dissolved UAl_4 is uniformly dispersed as in the case of the extruded material, than when segregated at the grain boundaries as in the cast slugs. (auth)

7649 CF-50-12-40
[Oak Ridge National Lab., Tenn.]
EXTRUSION AND HOT PIERCING OF URANIUM. R. J. Beaver. Dec. 7, 1950. Decl. Feb. 21, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Existing literature was surveyed in order to determine whether uranium can be extruded or hot pierced into tubing. (B.J.H.)

7650 CF-51-12-48
[Oak Ridge National Lab., Tenn.]
EXTRUSION CLADDING OF ZIRCONIUM TO URANIUM. R. J. Beaver. Dec. 4, 1951. Decl. Feb. 15, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Data are briefly summarized on the extrusion cladding of Zr to U. (B.J.H.)

7651 CF-52-12-109
Oak Ridge National Lab., Tenn.
SPOT WELDING OF STAINLESS CLAD FUEL ELEMENTS. G. M. Slaughter. Dec. 9, 1952. Decl. Feb. 14, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

The results of experiments on electric resistance welding of stainless steel clad fuel elements are reported. It seems probable that consistently sound, high-strength welds can be made if extremely precise control can be obtained. Suggested welding conditions are listed. (M.P.G.)

7652 CF-55-7-76
Oak Ridge National Lab., Tenn.
PERTINENT SPECIFICATIONS FOR FABRICATION OF

PLUTONIUM FUEL ELEMENTS. R. J. Beaver. July 20, 1955. Decl. Feb. 14, 1957. 9p. Contract [W-7405-eng-26]. $0.25(OTS).

Specifications for casting of Al – Pu alloy, breakdown of the casting into the plate assembly of the billet, and sealing of core within the billet are discussed. (J.E.D.)

7653 CF-55-8-114

Oak Ridge National Lab., Tenn.

INFORMATION OBTAINED FROM PROCESSING AMALGAMS PREPARED BY DRY SALT REDUCTION. E. S. Bomar. Aug. 22, 1955. Decl. Feb. 16, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

An examination of the behavior of cold pressed annealed amalgams prepared by reduction of dry salt was made and compared to that observed for amalgams from salt dissolved in PDA as an aid to deciding which method should be used in larger equipment. (F.S.)

7654 CT-422

Chicago. Univ. Metallurgical Lab.

TECHNOLOGICAL RESEARCH AND DEVELOPMENT, SECTION I. REPORT FOR MONTH ENDING JANUARY 15, 1943. Decl. Feb. 18, 1957. 37p. (A-437). $6.30(ph OTS); $3.00(mf OTS).

Results of progress on the casting, extrusion, thermal expansion, thermal transfer, workability, and tensile properties of uranium are reported. The microstructures of U containing small amounts of Mg and Ca and of C–U alloys containing 0.706 to 7.6% C are given. The casting of UC_2 and U_2C_3 is discussed. Operational methods and techniques being investigated in the production of U metal by Mg reduction of UF_4 are described. A thermocouple-controlled casting unit is described. The pyrophoricity of U metal powder was studied. Forging, rolling, and hardness tests on U specimens were made. (L.M.T.)

7655 CT-423

[Chicago. Univ. Metallurgical Lab.]

TECHNOLOGICAL RESEARCH, SECTION II, REPORT FOR MONTH ENDING JANUARY 15, 1943. Decl. Feb. 23, 1957. 24p. $4.80(ph OTS); $2.70(mf OTS).

Progress is reported in the following studies: experimental work in the production of Zn and other coatings for the protection of U; hot dipping, mechanical, and other methods for the application of chemical coatings to U; the effects of corrosion inhibitors and temperature on the corrosion of U and and preoxidized U by water; the corrosion of Cr, Fe–U alloy, U carbide and Si-coated U by Bi; the cathodic protection of U; the electroplating of U with Ni; the testing of coatings for U, including tests in boiling dilute H_2O_2, tests at high temperature in commercial He, the reaction of U with air, and corrosion of U in circulating systems. (C.H.)

7656 CT-468

Battelle Memorial Inst., Columbus, Ohio.

PROGRESS REPORT ON METALLURGY OF TUBALLOY TO UNIVERSITY OF CHICAGO. H. W. Russell. Feb. 1, 1943. Decl. Feb. 23, 1957. 32p. $4.80(ph OTS); $2.70 (mf OTS).

Hot extrusion of U in an experimental extrusion press produced satisfactory 3/8 in. rods, especially when quenched to reduce oxidation. The temperature used imposed severe requirements for press materials. Satisfactory U ingots were cast in a cold Cu mold. Observations are included on heating U in chloride baths. Vapor phase coating of U by Si and the electrodeposition of Ni were investigated and preliminary results are included. (C.H.)

7657 CT-541

Battelle Memorial Inst., Columbus, Ohio.

PROGRESS REPORT ON METALLURGY OF TUBALLOY TO UNIVERSITY OF CHICAGO. Mar. 10, 1943. Decl. Feb. 19, 1957. 38p. $9.30(ph OTS); $3.60(mf OTS).

Progress is reported in studies on the hot extrusion of U. Extrusion on a commercial scale was shown to be feasible. It was shown that protective coatings of sheet Al can be applied in a manner which will inhibit alloying and diffusion at temperatures above 1600°C. Metallographic studies on U billets revealed that the structure is dependent upon heat treatment. Data are included from tests on a number of die materials, liquid heating baths, Al coatings, Ni coatings, and procedures and materials used for the preparation of metallographic specimens. (C.H.)

7658 CT-688

Battelle Memorial Inst., Columbus, Ohio.

PROGRESS REPORT ON METALLURGY OF TUBALLOY TO UNIVERSITY OF CHICAGO. May 10, 1943. Decl. Feb. 19, 1957. 42p. $6.30(ph OTS); $3.00(mf OTS).

Progress is reported in the following studies: the amount of H contained in extruded U bars in equilibrium with H at one atm. pressure at various temperatures; the absorption of N by U at various temperatures; the effect of outgassing at 600°C for 1 to 3 hr in A atmospheres and in a vacuum on the behavior of Al-jacketed slugs; the cause of differences in machinability of U bars; the effects of B additions and annealing on the machinability of cast Fe; preliminary studies on flash welding of U; tests for hardness of U at elevated temperatures; and investigations of extrusion die materials. (C.H.)

7659 CT-818

Battelle Memorial Inst., Columbus, Ohio.

PROGRESS REPORT ON METALLURGY OF TUBALLOY TO UNIVERSITY OF CHICAGO. July 10, 1943. Decl. Feb. 18, 1957. 40p. Contract [W-7405-eng-92]. (A-1115). $7.80(ph OTS); $3.30(mf OTS).

Water tight tubes 18″ long have been centrifugally cast from beryllium, and tubes 30″ long have been cast. The hot extrusion of beryllium in the temperature range of 1000 to 1260°C does not appear feasible with present steels for the die and cylinder. Preliminary tests on flash welding of cast beryllium tubing indicate that water tight welds can be made in case it is necessary to join short lengths to produce the required length of tubing. The tests on aluminum jacketed slugs at 550°C have given failures after about 90 days caused by alloying through the jackets on those which have been longest on test. The metal hydrogen equilibrium as affected by temperature and pressure has been checked by more precise measurements causing some slight shifts in the curve previously reported, particularly in the high temperature region. A considerable number of hydrogen analyses have been made in investigating the analytical procedure used for hydrogen determination. Work now in progress seems to clarify these results. A simple method of hydrogen analysis has been developed. The results of hydrogen analyses in following a billet through the extrusion process show a decrease in hydrogen content in the bar and extrusion butt over that found in the billet. The butt end of bars shows two to three times the hydrogen content found in the leading end. An addition of lead in the metal does not appear to affect tool life on machining, but it appears to cause the chips to break up and reduces the tendency of the metal to burnish under the cut. Metallographic examination of all bars used on the machinability work shows no correlation of the microstructures on furnace-cooled specimens with machining characteristics. Hydrogen in the amounts greater than normally present in the metal has no apparent effect on the microstructure. Work in progress indicates that lamellar structure occurs in samples containing 200 to 400 ppm iron. (auth)

7660 CT-1388
Battelle Memorial Inst., Columbus, Ohio.
PROGRESS REPORT ON METALLURGY OF TUBALLOY TO UNIVERSITY OF CHICAGO. Feb. 10, 1944. Decl. Feb. 18, 1957. 33p. (A-2029). $4.80(ph OTS); $2.70 (mf OTS).

Progress is reported on the preparation of expendable steel sleeves for Al–Si bonded jackets by spinning steel tubing of the proper size. Present indications are that satisfactory sleeves can be made by this method. Data on the hydrogen solubility in U above the melting point have been added to those previously reported. These data show a hydrogen solubility at atmospheric pressure in the liquid of 28 to 30 ppm as compared to 16 to 17 ppm at 1000°C. A melt made in a CO atmosphere showed a reaction at the surface of the melt, producing a scale of oxide and carbide and an increase of carbide in the metal at the surface of the melt. There was no evidence of appreciable solubility. Aluminum plated with Ni or Cu and Cr after anodic treatment in phosphoric acid showed satisfactory wetting when hot-zinc dipped with flux, but very poor adherence. The Krome Alume Company's pretreatment gave greatly improved adherence, but wetting was not so satisfactory. There appears to be some relation between the amount of nickel or copper undercoat and the wetting characteristics which is being studied further. Diffusion tests at 190°C showed promising diffusion characteristics for the 10% Mg–Cu hot-dip coat when soldered with a 2% Sn–Pb solder. No diffusion of Pb was observed, and the amount of Sn diffusion, observed on this test, indicates a negligible amount at operating temperatures of 70 to 100°C. Additional scaling tests on U show that no scale is produced by the water quench in the temperature range 593 to 760°C. The rate of scaling in alpha and beta regions over a six-hour period shows no variations in rate indicative of changes in oxidation rate caused by different phases, but a direct relation with temperature. (auth)

7661 CT-2634
Chicago. Univ. Metallurgical Lab.
DIFFUSION OF ELECTROPLATES. Final Report on a Part of Problem Assignment No. 381-ML-54-3, F. S. 17. Section T-IV. F. Foote, J. R. Clark, B. J. Nelson, and H. Morris. Jan. 19, 1945. Decl. Feb. 19, 1957. 27p. Contract W-7401-eng-37. $4.80(ph OTS); $2.70(mf OTS).

U slugs have been electroplated with Ni, Cu over Ni, Cr over Ni and Mo with Ni. The electroplates were diffused by heating in rotating tube furnaces or by high fequency induction. The corrosion rates of these coated slugs was determined in synthetic Hanford-water at 90°C. The alloyed surface appeared to be attacked only slightly but localized corrosion at imperfections in the surface resulted in an over-all corrosion rate for 1000 hours which was only 10 to 100 times less than that of pure U. (auth)

7662 CT-2700
Battelle Memorial Inst., Columbus, Ohio.
METALLURGY OF TUBALLOY [URANIUM]. Progress Report. Feb. 1, 1945. Decl. Feb. 19, 1957. 50p. $7.80 (ph OTS); $3.30(mf OTS).

Data given on the properties of Th include those on thermal conductivity, annealing, pickling, oxygen content of extruded Th, and carbide formation. The properties of the Th–H system are also discussed. Phase changes in U–Nb and Ti–U alloys were studied. Data are included on corrosion testing of Ti–U alloys in H_2O. The melting and rolling of Al–U alloys is discussed. The thermal conductivity of U was studied. (B.J.H.)

7663 CT-2743
[Chicago. Univ. Metallurgical Lab.]
METALLURGY DIVISION REPORT FOR MONTH OF FEBRUARY 1945. Mar. 13, 1945. Decl. Feb. 20, 1957. 88p. Contract W-7401-eng-37. $13.80(ph OTS); $4.80 (mf OTS).

Tensile test data on heat-treated U samples are collected. Results of compression tests on U–Nb alloys and of microscopic and x-ray-diffraction studies on U–Nb alloys are also given. The casting of Al–U alloys is discussed. Corrosion data are given for Ta–U, V–U, and U-Zr alloys. The recrystallization of Th is reported, and some Al–Th alloys were examined. Results of Th coating experiments are discussed. Data are given on the corrosion of U in H_2O, NaCl solutions, and nitric acid. Corrosion data are also given for Mo alloys and U–Si systems. The corrosion of Th in H_2O is also discussed. (B.J.H.)

7664 CT-2787
[Chicago. Univ. Metallurgical Lab.]
METALLURGY DIVISION PROGRESS REPORT FOR MARCH 17 TO MARCH 24, 1945. F. Foote, E. W. Brugmann, and J. H. Chapin. Mar. 24, 1945. Decl. Feb. 16, 1957. 11p. Contract W-7405-eng-37. $3.30(ph OTS); $2.40(mf OTS).

Corrosion data on Nb–Zr–U, Nb–U, Be–Th, and Si–Th alloys in boiling distilled water are reported. The corrosive properties of liquid Na and BeF_2–NaF melts were investigated. Beryllium extrusion and thorium fabrication results are presented. Corrosion data on Be and Nb–U alloys in aqueous NaCl solutions are also reported. (C.W.H.)

7665 CT-2794
[Chicago. Univ. Metallurgical Lab.]
DIVISION OF METALLURGY REPORT FOR MONTH OF MARCH 1945. Apr. 12, 1945. Decl. Feb. 20, 1957. 110p. Contract W-7401-eng-37. $18.30(ph OTS); $6.00(mf OTS).

The casting of U into water-cooled Cu molds was investigated. Tensile tests on U–Nb and Mo–U alloys have shown that strength and ductility are markedly affected by heat treatment. Tantalum and Zr additions improved the corrosion resistance of uranium to boiling distilled H_2O. The hot-wire method is applicable for the production of small quantities of Th. The compressive strength of Th decreased rapidly with increasing temperature until 700°C was reached. Tensile properties, corrosion rates in H_2O, and phase studies of Th alloys are reported. Extruded Be rods showed internal defects. The corrosive properties of liquid Na were studied. Pretreatment of the Th surfaces had little effect on coating formation. Corrosion rates of Nb–U, Si–U, Mo–U, Al-U alloys and Be in boiling H_2O are reported. Work was continued on extrusion of Be and casting, rolling, and extrusion of Th. (C.W.H.)

7666 CT-2925
[Chicago. Univ. Metallurgical Lab.]
DIVISION OF METALLURGY REPORT FOR THE MONTH OF APRIL 1945. May 8, 1945. Decl. Feb. 23, 1957. 29p. Contract W-7401-Eng-37. $9.30(ph OTS); $3.60(mf OTS).

Investigations were continued on the corrosive effects of boiling distilled H_2O on Si–U, Mo–U, Nb–U, Th, and U alloys. Thorium has been refined by the hot wire method. A number of Be extrusions were examined for recrystallization. The corrosion of several metals in liquid Na at 500°C was studied. Attempts to plate Th by immersion plating of Zn followed by electroplating with Cu were not successful. Heat treatment of U improved its corrosion resistance to H_2O. Rolling and forging experiments were carried out on Th. (C.W.H.)

7667 CT-2950
[Ames Lab., Ames, Iowa.]
THE PRODUCTION OF THORIUM METAL BY THE METALLOTHERMIC REDUCTION OF THORIUM FLUO-

RIDE. F. H. Spedding, H. A. Wilhelm, W. H. Keller, James E. Iliff, and Charles Neher. Sept. 24, 1945. Decl. Feb. 16, 1957. 22p. Contract W-7405-eng-82. $4.80(ph OTS); $2.70(mf OTS).

A process is described by which large quantities of Th have been produced by the reduction of the UF_4 by Ca. It was found necessary to add heat to that of the reaction by carrying out simultaneously an auxiliary reaction in the reaction chamber. The reaction of iodine, bromine and various halides of volatile metals with Ca produced sufficient additional heat to fuse the slag and the metal. $ZnCl_2$ proved to be the most effective "booster." Conditions of reduction are described for various scales of operation. The properties of the resultant alloy and the purity of the final product are described. (auth)

7668 CT-3309
Massachusetts Inst. of Tech., Cambridge.
ALLOYS OF URANIUM AND SILICON. II. THE EPSILON PHASE. Gust Bitsianes. June 1945. Decl. Feb. 16, 1957. 32p. Contract W-7405-eng-175. $0.40(OTS).

Epsilon (ϵ) is an intermediate phase in the U–Si system containing about 23 at.% Si. This phase forms by means of a peritectoid reaction whose temperature is lowered by the presence of carbon. Investigation of the corrosion resistance of ϵ in various media shows that ϵ is quite superior to pure U metal and many other U alloys. The ϵ phase is a high-strength alloy and possesses some degree of ductility. These properties indicate a distinct possibility of using ϵ in direct contact with cooling water. (auth)

7669 FMPC-149(Rev.)
National Lead Co. of Ohio, Cincinnati.
SPECIFICATIONS FOR THORIUM PRODUCTION OPERATIONS. O. R. Magoteaux, E. W. Mautz, and T. C. Runion. Mar. 5, 1953. Decl. Mar. 18, 1957. 18p. Contract AT(30-1)-1156. $0.30(OTS).

Specifications, suggested vendors and estimated quantities of refractories, and chemical raw materials are presented as an aid to procurement for the thorium production plant. Suggested working specifications for $Th(NO_3)_4$ and thorium metal are also included. (auth)

7670 FMPC-332
National Lead Co. of Ohio, Cincinnati.
REPORT ON THE EVALUATION OF THE MGX SYSTEM OF THE FERNALD SAMPLING PLANT. James J. Costa. Oct. 6, 1953. Decl. Feb. 16, 1957. 24p. Contract AT(30-1)-1156. $0.35(OTS).

A comparison of weights, moistures, assays, and U_3O_8 content of representative test samples indicated good agreement between sampling methods employed at the Middlesex and Fernald sampling plants. (C.H.)

7671 HW-22975
[Hanford Works, Richland, Wash.]
EXPERIMENTAL ROLLING OF URANIUM—LACKAWANNA NO. 3. J. W. Riches. Dec. 13, 1951. Decl. Feb. 20, 1957. 7p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Uranium was machined after rough rolling and preheated at various temperatures in both lead and salt bath preheating furnaces. In the salt bath furnaces surface oxidation was reduced, rolling temperature was more easily controlled, and although the temperatures were as high as 593°C the specimens exhibited cold worked structure. (F.S.)

7672 ISC-219
Ames Lab., Ames, Iowa.
PROGRESS REPORT OF AN INVESTIGATION OF THE PROPERTIES OF THORIUM AND SOME OF ITS ALLOYS. G. C. Danielson, Glenn Murphy, D. Peterson, and B. A. Rogers. Apr. 1, 1952. Decl. Feb. 26, 1957. 23p. Contract W-7405-eng-82. $0.35(OTS).

7673 ISC-423
Ames Lab., Ames, Iowa.
QUARTERLY SUMMARY RESEARCH REPORT IN METALLURGY FOR JULY, AUGUST, SEPTEMBER 1953. Feb. 18, 1954. Decl. Feb. 26, 1957. 46p. Contract W-7405-eng-82. $0.40(OTS).

Progress is reported on the following investigations: niobite–tantalite ore treatment; separation of Nb from Ta and Nb and Hf from Zr by liquid-liquid extraction; preparation of Nd, Y, La, and Ce fluorides and $ThCl_4$; preparation of La, Ce, and V; preparation of U and Nb by reduction of their oxides; preparation of U–Zn and Th–Ni alloys; phase studies of Sn–Zr, Th–Zr, Al–V, Ta–V, C–Th, Nb–U, Nb–Zr, V–Zr, Mg–Th, and Zr–An alloys; pyrophoric film formed on U–Zr alloys pickled in HNO_3; Na–W bronze and $CaO–CaF_2$ system; hardness of as-arc-melted U–Ta alloy and Na–W bronze; fabrication of Nb–U alloys; corrosion of α-annealed and α quenched U–Zr–Nb alloy in boiling and 178°C water; corrosion of Nb and Ta in eutectic U–Cr alloy at 900°C; corrosion Zn dip-coated U in boiling H_2O and molten Al–Si system; casting U into Zr and U–Mb alloy cans; electroplating Zr cans and Zn dip-coated U with Cu and chrome plates; thermal cycling of Zr; and Zn-base and composite dip and diffusion coating of U in Mg bath containing Ni, Si, or Zr. (J.A.G.)

7674 ISC-425
Ames Lab., Ames, Iowa.
SEPARATION OF Ra^{228} IN THORIUM METAL PRODUCTION. D. T. Peterson, R. W. Moore, and M. Allison. Nov. 18, 1953. Decl. Feb. 26, 1957. 5p. Contract W-7405-eng-82. $0.20(OTS).

The separations of Ra^{228} which occur in the production of thorium metal by the Ames Process were studied by analyzing samples from various steps in the process. Approximately 70% of the Ra^{228} in the starting material remains in the filtrate in the precipitation of thorium oxalate. During the vacuum casting of the thorium metal, approximately 50% of the remaining Ra^{228} is removed. No significant separations were found in the other steps in the process. (auth)

7675 ISC-568
Ames Lab., Ames, Iowa.
PURIFICATION OF URANIUM BY SOLVENT EXTRACTION. Dean E. Griffith and Morton Smutz. Jan.13, 1955. Decl. Feb. 26, 1957. 38p. Contract W-7405-eng-82. $0.40(OTS).

A product containing at least 99.5% uranyl nitrate was obtained from the U concentrate produced using the Ames monazite process in pilot plant work at the Ames Laboratory. The U concentrate contained only about 1% U. The major constituents were rare earth phosphates and sulfates. The purification steps consisted of dissolving the concentrate in 6.25N HNO_3, extracting the U from the rare earths using undiluted TBP as a solvent in a multistage extractor, stripping the U from the solvent, and precipitating impurities with $H_2C_2O_4$. (auth)

7676 ISC-575
Ames Lab., Ames, Iowa.
QUARTERLY SUMMARY RESEARCH REPORT IN METALLURGY FOR OCTOBER, NOVEMBER, DECEMBER 1954. July 14, 1955. Decl. Feb. 27, 1957. 39p. Contract W-7405-eng-82. $0.40(OTS).

The separation of several metals from Cr–U alloys by liquid metal extraction, the separation of Nb from Ta, and other purification studies were investigated. Studies on the preparation of various metals and metallic halides were continued. The effect of carbon on the electrical re-

sistivity of Th was measured. Phase studies were reported for the U–Zn, U–Nb, and Ta–Zr alloys. The corrosion of refractory metals and metallic carbides by liquid U and U–Cr alloys was investigated. The elastic constants of lead crystals and the electrical resistivity and diffraction patterns of Ca allotropes were determined. The structures of Zr–Ge and Zr–U alloys were investigated. The corrosion resistance of Zr was studied. Several other investigations are outlined. (For preceding period see ISC-531.) (C.W.H.)

7677 ISC-607(Del.)
Ames Lab., Ames, Iowa.
QUARTERLY SUMMARY RESEARCH REPORT IN METALLURGY FOR JANUARY, FEBRUARY, MARCH 1955. Oct. 21, 1955. Decl. Feb. 26, 1957. 37p. Contract W-7405-eng-82. $0.40(OTS).

Studies on the separation of Nb from Ti, Hf from Zr, and Mg from Ca by sublimation and metals from Cr–U eutecti by liquid metal extraction were continued. Methods for the preparation of $ThCl_4$, YF_3, V, Ta, Nb, Th, and Y are presented. A study of the following alloys is reported: U–Zr, Th–Zr, Th–Hf, Th–C, Nb–U, Ta–Zr, Zr–Zn, Ta–V, and Nb–Zn. Corrosion of refractory and coated metals by liquid metals, carbides by molten U, hot-press bonding of Al to other metals, thermal cycling of metals through their transformation temperatures, the structure of Ge–Zr compounds, and corrosion of H_2SO_4 treated Zr and zircaloy in 300°C water were investigated. (J.E.D.)

7678 KAPL-321
Knolls Atomic Power Lab., Schenectady, N. Y.
REACTOR MATERIALS–SECTION III [OF] PROGRESS REPORT NO. 44 [FOR] MARCH 1950. Apr. 20, 1950. Decl. Feb. 26, 1957. 28p. Contract W-31-109-Eng-52. $4.80(ph OTS); $2.70(mf OTS).

Reactor Materials. Coarse-grain specimens of alpha uranium were prepared and deformed for measurement of the angles between the traces of twins and between the traces and specimen axes. The crystallographic data necessary for interpreting such measurements, namely, standard stereographic projections of important index planes, were prepared. Gas-lubricated pistons for loading wire specimens into a reactor to study creep under irradiation were made and studied primarily to determine behavior and leak rate as a function of clearance between the piston and the cylinder. A linear differential transformer was calibrated and is being tested for use in conjunction with this creep apparatus. A Guoy balance for measuring magnetic susceptibility in weakly magnetic metals was completed. Specimens of type-347 stainless steel tested to date show too much magnetism for the employment of such a method. Attempts were made to rid the specimen of ferritic material. Beta Experiment. The first 5 sputtering foils from the 5% burnup titanium beta experiment pin were analyzed for fission products, uranium, and titanium. The first 3 foils contained virtually no fission-product activity or uranium. The fourth foil (obtained before the pin punctured) contained a little activity as well as uranium. The fifth foil (punctured pin) contained considerable fission-product activity and a relatively large amount of uranium. Analytical Studies. In order to provide rapid control analyses for the Genie Program, a method for the determination of calcium was developed which depends on the separation and concentration of calcium on an ion-exchange column. Because of the importance of trace impurities in the structural materials used in the WMA Reactor from a nuclear point of view, it became necessary to devise sensitive and accurate methods for the determination of these impurities in stainless steel. Accordingly, methods were developed for the determination of those impurities (cobalt, silver, tantalum, cadmium, and boron) with a relatively high cross section. Reduction of VF_3 with Hydrogen. A preliminary experiment was run to determine if VF_3 can be reduced to vanadium metal by hydrogen. This reaction is thermodynamically possible, and a small amount of vanadium metal was produced as indicated by x-ray analysis. Th Analysis of Nitride Nitrogen in Vanadium. Nitrogen which was present in vanadium metal as vanadium nitride posed a very difficult analytical problem. Recent experiments indicated that this nitrogen is released as molecular nitrogen when fused with potassium pyrosulfate. A new technique was worked out in which molten lithium sulfate is used as the medium for the pyrosulfate oxidation. Vanadium heated in nitrogen was examined microscopically. After 1.5 hr at 1400 to 1450°C, nitrogen diffused several millimeters into the vanadium, as well as forming a layer on the surface. Microscopic evidence showed that a nitride was precipitated on cooling of the specimen in the interior. The nitride appeared to be isomorphous with the carbide. With the installation of equipment in new facilities, transformation kinetics in uranium alloys of interest to Hanford, containing about 0.6 at. % chromium, were resumed. As a part of a study on fabrication of uranium and uranium–chromium alloys, thermal cycling equipment was put into operation. With the resumption of work in corrosion testing in new quarters, the testing of protective coatings on beryllium is being carried on. Shipboard-reactor fuel elements containing BeO and UO clad in stainless steel are being studied by swaging stainless-steel tubes filled with MgO or graphite (used as stainless), and suitable densities were achieved by proper reduction of area of the tube assembly. The use of lithium as a control monitoring material is being explored. Corrosion programs are listed. A production run of 300 ft of titanium capsule tubing is being studied carefully. Radiographic equipment and a Sperry ultrasonic reflectoscope have been put into operation for inspection of fuel rods. Preparations for testing irradiated Type-302 stainless-steel, Inconel, and Inconel-X springs are in readiness for material which will be sent from Hanford. (auth)

7679 MCW-29
Mallinckrodt Chemical Works, St. Louis.
A STUDY OF THE EFFECT OF VARIOUS FACTORS ON THE YIELDS OBTAINED IN THE REMELTING OF URANIUM METAL. W. A. Oppold. Apr. 1, 1946. Decl. Feb. 27, 1957. 19p. Contract W-7405-eng-29. $3.30(ph OTS); $2.40(mf OTS).

Remelt yield is the ratio of the weight of poured metal to the weight of the charge, which was produced by Mg reduction of UF_4 in a dolomite-lined steel vessel. In order to get the highest remelt yield consistent with quality, the effects of N_2 inclusion in the Mg used for reduction, Mg content of the dolomite lining, degree of vacuum in the remelt furnace, inert gas atmospheres, pouring temperature, and condition of the graphite crucible have been investigated. A MgO dressing is recommended for reducing pitting and lengthening the life of the crucible. (G.Y.)

7680 MIT-1109
Massachusetts Inst. of Tech., Cambridge.
[Metallurgical Project.]
IRON AS A CORROSION INHIBITOR IN BERYLLIUM WITH NOTES ON VECTORIAL CORROSION IN SINGLE CRYSTALS. C. S. Pearsall. Apr. 1953. Decl. Feb. 25, 1957. 19p. Contract AT(30-1)-981. $0.30(OTS).

The vectorial nature of corrosion in Be single crystals was studied. The effect of Al, Fe, and Si impurities on the corrosion behavior of Be is reported. (J.E.D.)

7681 MIT-1115
Massachusetts Inst. of Tech., Cambridge. Metallurgical Project.
TECHNICAL PROGRESS REPORT FOR THE PERIOD

OCTOBER 1953 THROUGH FEBRUARY 1954. May 4, 1954. Decl. Feb. 26, 1957. 138p. Contract AT(30-1)-981. $0.75(OTS).

Beryllium sheet having a 30 to 40% tensile elongation in all directions in the plane of the sheet has been produced by extrusion and cross rolling of powders. The solubility of U and of Ru in liquid Al, Mg, Sn, Pb, and Zn has been studied. Ruthenium is much more soluble in Mg than is U. A minimum in the thermal cycling growth of alpha extruded U-Zr alloys occurs at a Zr content of about 0.4 to 0.8 wt.% Zr. This favorable result occurs simultaneously with the appearance of a coarse grain size and might be different if the grains were fine. Zirconium apparently is more resistant than Ti to attack by molten U-Cr alloy. Casting of U-Cr alloy around clean Fe tubes which are internally cooled with molten Pb does not appear to be a feasible method of making the Dow-Detroit radiator. The best selection of temperatures for the incoming U-Cr and the Pb coolant led to attack of the outer layers of tubes while the center of the radiator remained unfilled. Various problems encountered in the making of Al-U alloy sandwich tubes with sealed ends are described. The uniformity of Zr-clad thickness on coextruded Zr-clad U has been studied for various alloys and prior treatments of the uranium. Previously extruded U-1.3 wt.% Zr alloy gives the best results, and the 0.16 wt.% Si alloy looks promising. The pressures required to extrude metals at various temperatures have been compared with the true stress-true strain tensile results at the same temperatures. (For preceding period see MIT-1114.) (auth)

7682 MonN-146

[Clinton National Lab., Oak Ridge, Tenn.]

CANNING OF THORIUM CARBONATE. Progress Report. Problem Assignment No. TX5-12. S. E. Beall and O. Sisman. July 30, 1946. Decl. Mar. 2, 1957. 12p. Contract W-35-058-eng-71. $3.30(ph OTS); $2.40(mf OTS).

The development of a procedure for the canning of thorium carbonate to be bombarded in the ORNL Graphite Reactor is described. (auth)

7683 NMI-1126

Nuclear Metals, Inc., Cambridge, Mass.

TECHNICAL PROGRESS REPORT FOR THE PERIOD JULY 1954 THROUGH SEPTEMBER 1954. Jan. 19, 1955. Decl. Mar. 4, 1957. 132p. Contract AT(30-1)-1565. $0.75(OTS).

A careful study has been carried out to look for the reasons why extruded cast beryllium has much less ductility than extruded powder beryllium of the same analysis. Water quenching at the extrusion exit produces as perfect a texture in cast beryllium as is ordinarily found in powder but the ductility of the former is still low. The grain size of extruded cast is larger than in extruded powder apparently because oxide films inhibit growth in the latter. Extrusion of a series of particle sizes proved that the final grain size is largely determined by the initial particle size. The poorer ductility of coarser grained material having a good texture is apparently caused by differences in c axis orientation (about the rod axis) of neighboring grains. This type of orientation difference can be eliminated by cross rolling an extruded flat. In this condition it appears that fabricated cast beryllium can be made as ductile as cross-rolled powder. This may make it unnecessary to use powdered beryllium for high ductility material. The best corrosion resistance for U-Si epsilon alloys appears to occur at about the U_3Si composition. The material can be extrusion clad with zirconium at about 850°C. The bond appears to be mechanically strong. It has not yet been determined that bond layer corrosion can be avoided. The clad epsilon can be cold swaged to a limited extent. Promising results have been obtained with extrusion cladding of thorium. It is necessary to use relatively soft thorium to obtain a uniform thickness of the aluminum clad. (auth)

7684 NMI-1145

Nuclear Metals, Inc., Cambridge, Mass.

THE URANIUM-SILICON EPSILON PHASE. S. Isserow. Apr. 20, 1956. Decl. Feb. 27, 1957. 65p. Contract AT(30-1)-1565. $0.50(OTS).

The U-Si epsilon phase was investigated for fuel element application in a water environment. The corrosion behavior was emphasized. Bare samples were tested in 500 and 650°F water and in 750°F steam. Evidence is presented that the corrosion resistance depends on the absence of phases other than epsilon and that the desired epsilon phase has a Si content in the range of 3.9 to 4.0 wt. %, rather than the 3.78 wt. % of stoichiometric U_3Si. It was necessary to develop an analytical method to overcome the apparently low Si content found in this system under certain conditions. The epsilon alloy was clad by coextrusion with Zircaloy-2. At 1560°F, reductions of 40X were readily achieved. Samples from such coextrusions, notably simulated fuel elements with intentional defects, were tested in 650°F water. Interaction between core and cladding was found responsible for corrosion along the interface, leading to failure of defected rods. Other properties of the alloy, including irradiation behavior, are briefly discussed. (auth)

7685 NYO-1129

Sylvania Electric Products Inc., Bayside, N. Y.

HOT PRESSING OF URANIUM. N. P. Pinto and H. H. Hausner. July 8, 1950. Decl. Feb. 27, 1957. 23p. Contract AT-30-1-GEN-366. $0.35(OTS).

Solid U of 18.7 to 18.8 g/cc density can be manufactured by hot pressing U powder at a compacting pressure of approximately 30 tsi and a temperature of approximately 600°C. By using U powder made from UH_3, solid U parts of rather unique properties can be produced. Since U parts made by this method are processed exclusively in the alpha range of U, they contain only small alpha grains of random orientation. Ordinary methods of manufacturing U by melting and subsequent rolling, extruding, or swaging, and also by ordinary powder metallurgy methods are discussed. The physical properties of this ordinary type of U are compared with those of hot-pressed U. The process of hot pressing is described in detail. The die design and die material necessary for this hot-pressing method are discussed. The feasibility of large-scale production of U parts of various dimensions by the hot-pressing method is reviewed. (auth)

7686 NYO-3680

Horizons, Inc., Cleveland.

INVESTIGATIONS FOR THE PRODUCTION OF THORIUM METAL. Technical Progress Report [for] Seventh Quarter, December 1, 1953 to February 28, 1954. Jason C. Bleiweiss and Bertram C. Raynes. Feb. 26, 1954. Decl. Feb. 27, 1957. 23p. Contract No. AT(30-1)-1335. $4.80 (ph OTS); $2.70(mf OTS).

An electrolytic process for producing Th metal, including the preparation of anhydrous $ThCl_4$, the electrolysis of this salt in molten NaCl, and the recovery of Th powder, are reported. The design of the cell used in the electrolytic process is given. (For preceding period see NYO-3730) (J.E.D.)

7687 NYO-3726

Horizons Inc., Cleveland.

INVESTIGATION FOR THE PRODUCTION OF THORIUM METAL. Technical Progress Report—Second Quarter, August 1—October 31, 1952. Merle E. Sibert, F. C. Wagner, and Morris A. Steinberg. Dec. 3, 1952. Decl.

Feb. 27, 1957. 25p. Contract AT(30-1)-1335. $4.80(ph OTS); $2.70(mf OTS).

Progress of the investigation concerning the preparation of thorium metal by electrolytic procedures is reported. This work has thus far been confined to electrolysis of thorium double halides in alkali halide media. The electrolysis of $KThF_5$ in NaCl has been successful, but the salt bath has a limited life. The use of $NaThCl_5$ has also been successful but not consistently so. This latter procedure is of considerable interest due to its possibilities for use as a continuous process. Operating data for these and other systems are given. Work was initiated on obtaining decomposition potentials of salts which show promise as source materials. Preliminary results are included. Some physical metallurgical evaluation of the metal produced has also been accomplished. This is reported along with pertinent analytical data. (auth)

7688 ORNL-642
Oak Ridge National Lab., Tenn.
URANIUM SAMPLES FOR PILE RADIATION EXPOSURE. A. G. H. Andersen. Mar. 31, 1950. Decl. Mar. 1, 1957. 40p. Contract W-7405-eng-26. $0.40(OTS).

Three ORNL reactor slugs and two uranium ingots enriched with U^{235} to 6.5 and 2.7%, respectively, were worked into square wires. The ingots were α-rolled into round rods, α-swaged, squared in a rolling mill, and finished to size by drawing through square dies. (auth)

7689 ORNL-1633(Del.)
Oak Ridge National Lab., Tenn.
FABRICATION OF SPHERICAL PARTICLES. E. S. Bomar, H. Inouye, and Art Levy. Dec. 8, 1953. Decl. Mar. 6, 1957. 12p. Contract W-7405-eng-26. $0.25 (OTS).

Several potential methods for the preparation of spherical particles of alloys containing U have been examined. In each instance, surface tension in the liquid state has been employed as the force for formation of spheroids. Spraying from a metallizing gun or forced ejection of molten alloy from a hole in the bottom of a crucible gave the most positive results. Both these methods, however, produced a wide range of particle sizes. (auth)

Refer also to abstracts 7697 and 7884.

PHYSICS

7690 ANL-5175
Argonne National Lab., Lemont, Ill.
PHYSICS DIVISION SUPPLEMENT TO QUARTERLY REPORT [FOR] SEPTEMBER, OCTOBER, AND NOVEMBER 1953. Feb. 1954. Changed from SECRET Nov. 29, 1955. Decl. Feb. 12, 1957. 27p. Contract W-31-109-eng-38. $4.80(ph OTS); $2.70(mf OTS).

Experimental Nuclear Physics. An investigation of the isomeric transition associated with Am^{242m} failed to reveal any L-conversion lines for either E3 or M1 transitions. New data are given for the total neutron cross sections (kev region) for Li^6, Li^7, Th, U, U^{235}, and Pu. Mass Spectroscopy. Values of α for U and Pu in the EBR have been determined for extended points within the core and blanket. Experimental Reactor Physics. Initial design factors for the source reactor of the Fast Exponential Experiment are outlined. Four irradiated NRX fuel rods were analyzed for change in reactivity. (K.S.)

7691 BNL-236
Brookhaven National Lab., Upton, N. Y.
QUARTERLY PROGRESS REPORT [FOR] JANUARY 1–MARCH 31, 1953. Decl. Feb. 13, 1957. 38p. $6.30(ph OTS); $3.00(mf OTS).

Progress is reported on the following studies: critical buckling experiments as a function of H_2O-to-metal ratios for 1.3% U^{235} fuel rods; determination of the fast-fission factor for $U-H_2O$ lattices; intracell flux distribution measurements for two lattices; preliminary design studies of the U–Bi liquid metal fuel reactor and the U–Pb–Sn fuel system; theoretical calculations on the accuracy of buckling measurements and reflector savings; relaxation methods for calculating average moderator flux; recalculation of ϵ for $U-H_2O$ lattices with new fast fission cross-section data; measurement of stored energy in the graphite of the BNL Reactor; Liquid Metal Fuel Reactor fuel processing and continuous breeding studies; heat-removal studies in a ThF_4 breeding blanket; creep and viscosity investigations of fused KCl–LiCl eutectic for heat removal; modification of the external cooling system of the LMFR to permit smaller fuel inventories; Th–Bi–Pb system phase studies; production of fine Th–Bi dispersions in liquid Bi and Pb–Bi eutectic; static corrosion testing of Fe–Mo alloys and croloy; liquid–metal convection phenomena; initial solubility studies of Fe, Cr, and Zr in liquid Bi; in-pile creep characteristic measurements on 2S Al; vacuum annealing of pure Fe; measurement of radiation damage on pile graphite and SiO_2; metallographic analysis of Fe–Mo alloys; slug element thermal cycling; neutron streaming in C steel. (K.S.)

7692 ISC-130
Ames Lab., Ames, Iowa.
QUARTERLY SUMMARY RESEARCH REPORT FOR JULY, AUGUST AND SEPTEMBER 1950. W. E. Dreeszen. Nov. 15, 1950. Decl. Feb. 28, 1957. 42p. Contract W-7405-eng-82. $7.80(ph OTS); $3.30(mf OTS).

Progress is reported on the metallurgy of Zr, the separation of rare earths by ion exchange, the determination of acid constants of dibasic acids, spectroscopic methods for the analysis of the rare earths, theoretical nuclear physics, β and γ spectroscopy, other experimental nuclear physics, fission fragment ionization measurements, and chemical engineering. (B.J.H.)

7693 NAA-SR-251
North American Aviation, Inc., Downey, Calif.
SOLID STATE AND IRRADIATION PHYSICS QUARTERLY PROGRESS REPORT [FOR] JANUARY – MARCH 1953. E. C. Crittenden, Jr. and F. E. Faris, eds. Oct. 8, 1953. Decl. Feb. 7, 1957. 63p. Contract AT-11-1-Gen-8. $10.80(ph OTS); $3.90(mf OTS).

Fundamental studies on pile-irradiated graphite which are reported include thermoelectric power at low temperatures, annealing mechanism, outgassing, release of stored energy, and thermal expansion properties. Data are presented showing the annealing of interstitial-vacancy pairs by fission fragments in Th and Th–Au alloys when bombarded by protons and deuterons. Copper specimens were cold worked at –196°C, and changes in the elastic modulus, resistivity, and internal friction after pulse annealing were observed. Changes in the elastic modulus of cold-worked Al after pulse annealing are also reported. Preliminary results are reported on the transformation mechanism in Cu–Fe alloys aged at 700°C for 24 hr. Further information is provided on the effects of radiation on the ordering-disordering rates in Cu_3Au specimens. Other fundamental studies on Cu include thermoelectric properties of anneal-damaged specimens, change in resistivity as a function of elastic tension, hardness changes when irradiated with 1.25-Mev electrons, displacement energy of electron-irradiated specimens, and effect of cold work on magnetic susceptibility. The growth of the Z band in KCl under x-

ray bombardment was observed. The fraction of vacancy pairs in NaCl dissociated at various temperatures is calculated. Preliminary results on the in-pile measurement of the thermal conductivity of U-impregnated graphite are reported. Results of optical absorption measurements on fluorite crystals irradiated with electrons are discussed. (L.M.T.)

7694 NBS-D-123

National Bureau of Standards, Washington, D. C.
QUARTERLY PROGRESS REPORT FOR JANUARY, FEBRUARY AND MARCH 1953. June 1953. Decl. Feb. 28, 1957. 59p. $9.30(ph OTS); $3.60(mf OTS).

A new method of construction and open-circuit voltage tests are reported for Zamboni piles (high-potential batteries). Evaluation of Jordan survey meters and Taplin liquid dosimeters is briefly reported. The method of determining the stored energy of irradiated graphite is described, and the effects of prolonged annealing periods on the stored energy are discussed. The synthesis of maltose-1-C^{14} is reported. Work on the preparation of Be dialkyls, Be diaryls, and BeI_2, and on the synthesis of various silanes is discussed. Additional phase studies are reported for the systems U–Ag, U–Au, U–C, UC–UC_2–Be_2C, and UC–Be_2C. (L.M.T.)

7695 UCLA-379

California. Univ., Los Angeles. Atomic Energy Project.
QUARTERLY PROGRESS REPORT FOR PERIOD ENDING SEPTEMBER 30, 1956. Oct. 1, 1956. Decl. Mar. 6, 1957. 148p. Contract AT-04-1-GEN-12. $22.80(ph OTS); $7.20 (mf OTS).

Progress is reported in the following studies: the synthesis and analysis of unsaturated fatty acids; the effects of whole-body x irradiation on the blood plasma Fe level in rabbits; an evaluation of the effect of quinoxaline 1:4-di-N-oxide on the survival time of irradiated mice; the effect of acute and chronic intravenous infusions of Dextran in irradiated rabbits; investigations of tracer techniques in liver function tests, kidney function tests, and thyroid function tests; the effect of pre-treatment with sodium salicylate on radiation mortality in mice; the skeletal deposition of Y^{91}, Ca^{45}, and S^{35} in rats following bone fracture; factors affecting the gastrointestinal absorption of Sr^{90}; the causative organisms in mouse bacteremias induced by x irradiation; the radiosensitivity of tobacco mosaic virus; effects of irradiation on lipoproteins in embryonic serum; routine monitoring and waste disposal activities; the analysis of human bone samples for Sr content; improvements in the performance of electron microscopes; the performance of chemical radiation detectors when used as dosimeters; factors affecting the uptake of fission products by plants and animals; the accumulation and persistence of radioactive fall-out material in samples of soils, vegetation, animal tissues, and milk collected at various sites and at various times; and a review of soil sampling and processing procedures for radiometric analysis. (For preceding period see UCLA-371.) (C.H.)

INSTRUMENTS

7696 CF-55-7-53

Oak Ridge National Lab., Tenn.
MOORE PMD REPORT. A. G. Grindell. July 8, 1955. Decl. Feb. 14, 1957. 22p. Contract [W-7405-eng-26]. $0.35(OTS).

A complete description is given of the latest Moore Products Company null-balance pressure transmitter, embodying a 3-ply Inconel bellows and Inconel body, with a 100 psi range. Tests have been made up to 1200°F, in liquid metal and fused salt systems. Blueprints are included. (B.J.H.)

7697 CT-2633

[Chicago. Univ. Metallurgical Lab.]
SUPERSONIC FLAW DETECTOR, MODEL 1 T. Final Report on Part of P. A. No. 154-ML-54-3. Robert F. Plott. Apr. 7, 1945. Decl. Feb. 18, 1957. 47p. Contract W-7401-eng-37. $0.45(OTS).

Results of an investigation of supersonic waves as a means for the nondestructive detection of flaws in Hanford slugs are reported. These results were compared with standard detection methods, autoradiography, and frost testing. The comparison indicates the supersonic detector is a valuable tool in the determination of bond perfection. (D.E.B.)

7698 DP-60

Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Augusta, Ga.
TRANSFORMATION AND NON-BOND TESTER FOR FLAT ELEMENTS. W. L. Blucke. Apr. 1954. Decl. Feb. 20, 1957. 35p. Contract AT(07-2)-1. $0.35(OTS).

A composite ultrasonic unit was developed to indicate the size and location of nonbonded areas and the presence of nontransformed regions in flat uranium fuel elements clad with aluminum. A two-dimensional recorder was developed to yield instantaneous, permanent records of the nonbonded areas on electrosensitive paper. Panel lights indicate the presence of nontransformed regions that have not received proper heat treatment. Defects $\frac{1}{8}$ in. in diameter are consistently detected at an inspection rate of approximately 50 $in.^2$/min. (auth)

7699 DP-68

DuPont de Nemours (E. I.) & Co. Savannah River Lab., Augusta, Ga.
A DETECTOR FOR SUB-SURFACE CRACKS. L. E. Goodwin. Aug. 1954. Decl. Feb. 20, 1957. 24p. Contract AT(07-2)-1. $0.30(OTS).

An instrument was developed to detect cracks within $\frac{1}{8}$ in. of the surface of cylindrical U fuel slugs. The instrument measures changes in eddy currents that are induced at a frequency of 2 kc. (auth)

7700 KLX-1361

Vitro Corp. of America, New York.
MAGNETIC INDUCTION FLOWMETER DEVELOPMENT. JOB 24-B4. Project Status Report for the period May 1, 1950–June 30, 1951. Oct. 16, 1952. Decl. Feb. 7, 1957. 75p. Contract AT(30-1)-850. $0.55(OTS).

A magnetic induction flowmeter for use in radiological and radiochemical plant operations was developed. Prototype models in three flow ranges, 0 to 1,000 cc/min, 0 to 3 gpm and 0 to 200 cc/min were built and tested in the Vitro laboratory. Initial service testing of the 0 to 1000 cc/min model at ORNL was completed. A program for the work necessary to refine the present flowmeter design was outlined. Additional models are being built for other service tests at various AEC installations and for life tests in the Vitro laboratory. (auth)

ISOTOPE SEPARATION

7701 A-531

Columbia Univ., New York. Div. of War Research.
THEORETICAL CONSIDERATIONS ON LIQUID THERMAL DIFFUSION EXPERIMENTS. Karl Cohen. Feb. 5, 1943. Decl. Feb. 12, 1957. 22p. Contract OEMsr-412. (4-R-21). $4.80(ph OTS).

7702 A-779
Yale Univ., New Haven. Sloane Physics Lab.
FINAL REPORT ON WORK UNDER O. S. R. D. Contract OEMsr-911. William W. Watson. Aug. 5, 1943. Decl. Feb. 12, 1957. 4p. (100XR-1895). $1.80(ph OTS); $1.80 (mf OTS).

Experiments designed to indicate the feasibility of B^{10} concentration by thermal diffusion using BF_3 are discussed. B^{10} may be concentrated by this method, but with very low efficiency. (D.E.B.)

7703 A-854
National Bureau of Standards, Washington, D. C.
PROGRESS REPORTS ON MOLECULAR DISTILLATION FROM FEBRUARY 10, 1943 TO MAY 25, 1943. A. Keith Brewer, Samuel L. Madorsky, and T. Ivan Taylor. Decl. Feb. 14, 1957. 39p. $6.30(ph OTS); $3.00(mf OTS).

Progress in the development of isotope separation by molecular distillation is reviewed. (D.E.B.)

7704 CF-54-11-162
Oak Ridge National Lab., Tenn.
FRACTIONATION OF URANIUM ISOTOPES BY ION EXCHANGE. J. S. Drury and A. C. Rutenberg. Nov. 23, 1954. Decl. Feb. 14, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

An experiment was performed to determine the isotope fractionation obtained by passing a band of U through an ion exchange column. The conditions of the experiment were similar to those existing in the experiment described in ISC-475, except that an 11-in. band of U was passed through 64 in. of Cu resin. Very little enrichment resulted. (auth)

7705 MonT-155
[Clinton Labs., Oak Ridge, Tenn.]
SUMMARY OF 706-D Ba^{140} PRODUCTION RUNS 1-12. Problem Assignment TX1-1. W. A. Rodger. Aug. 30, 1946. Decl. Feb. 27 1957. 13p. Contract W-35-058-eng-71. $3.30(ph OTS); $2.40(mf OTS).

Data are summarized on equipment changes made during dummy and active runs for the production of Ba^{140} following separation by wet precipitation. (C.H.)

MASS SPECTROGRAPHY

7706 A-1987
[National Bureau of Standards, Washington, D. C.]
AN AUTOMATIC RECORDING SYSTEM FOR THE URANIUM MASS SPECTROMETER. A. Keith Brewer, Roy J. Britten, and Paul Bradt. [Sept. 30, 1944]. Decl. Feb. 12, 1957. 9p. $0.25(OTS).

Circuit diagrams are given for the automatic recording system, and its operation is briefly discussed. (B.J.H.)

NUCLEAR PHYSICS

7707 ANL-4284
Argonne National Lab., Lemont, Ill.
THEORETICAL NUCLEAR PHYSICS DIVISION. Report for Period Ending March 28, 1949. May 2, 1949. Decl. Feb. 11, 1957. 61p. Contract W-31-109-eng-38. $9.30(ph OTS); $3.60(mf OTS).

A mathematical analysis is made of neutron attenuation in a conical shield. An analysis is also made of the efficiency of a collection chamber for Br ion collection as a function of the charge, mass, and energy of the Br nuclei, the radii of the outer and inner cylinders of the chamber, and the potential difference between the cylinders. Information on the shell structure of nuclei is summarized. Calculations were made by the Monte Carlo method on the neutron distributions in two fast reactors, the Experimental Breeder Reactor and the Los Alamos Fast Reactor. Results are given in both tabular and graphical form. Calculations were also carried out on a proposed research reactor for the Argonne National Laboratory. Included are studies of two-group constants, critical mass and flux distributions, and adjoint fluxes. (B.J.H.)

7708 ANL-4757
Argonne National Lab., Lemont, Ill.
NEUTRON PRODUCER (CP-6) PROGRAM TECHNICAL PROGRESS LETTER; JANUARY 1952. Stuart McLain, comp. Feb. 12, 1952. Decl. Feb. 12, 1957. 35p. Contract W-31-109-eng-38. $6.30(ph OTS); $3.00(mf OTS).

Rapid U oxidation in water was studied in a pressurized section of a loop containing a slug and heated internally by a calrod unit. The loop section if shown as well as the exposed section of the slug after oxidation and corrosion. Simultaneously with the loop tests, UO_3 solubility tests were also explored and reported. The decomposition of heavy water in the CP-3 has been measured for a He gas flow of 7 cfm. The corrosion test results are tabulated for Al and Al-Si specimens. Preliminary and first-phase tests of both the hydraulically operated bundle-type control rod and the hydraulically operated safety rod are reported. Out-of-roundness and low warping measurement of α-canned U slugs are reported. Analytical results are tabulated for the hourly values of U losses in the aqueous waste steam of a box extractor. The results of a differential distillation of 800 ml of synthetic IAW solution spiked with radioactive Zr are tabulated and graphed. (C.J.E.)

7709 BNL-225
Brookhaven National Lab., Upton, N. Y.
QUARTERLY PROGRESS REPORT FOR OCTOBER 1–DECEMBER 31, 1952. Decl. Feb. 13, 1957. 34p. $6.30 (ph OTS); $3.00(mf OTS).

Progress is reported on the following studies: measurements of reactor constants for lattices of 0.600- and 0.750-in.-diam. 1.0 and 1.3% U^{235} rods in H_2O moderator for CVR; LMFR design; B–U^{233} core and external ThF_4 blanket; dispersion of an intermetallic U compound in liquid Pb–Sn; coolants of Bi–Th and PB–Bi–Th; heat-transfer properties of H_2O–U rod lattices at high Reynolds numbers; local heat-transfer coefficient data; table of reflector savings in H_2O–U lattices by two-region, two-group calculations; minimum critical mass for 1.3% U–H_2O lattice; variation of σ_{tr} of U with energy from 0.1 to 10 Mev; design and construction of fused-salt extraction in-pile and out-of-pile loops for LMFR; removal of U^{233} from compressed spheres of ThF_4 in reactor; sintering and regeneration of ThF_4; continuous dissolver unit for BrF_3 process; studies of pilot plant for packaging Savannah River wastes; phase studies on the systems U–Bi, U–Pb–Bi, U–Pb–Sn, and Th–Bi–Pb; dispersions of $ThBi_3$ in liquid Bi and Th_3Bi_5 in liquid Pb–Bi; design for reactor employing metallic dispersion as liquid metal fuel; study of U–Bi–Fe and U–Bi–Ni systems with relation to container problem; in-pile creep testing; further study of radiation damage to SiO_2; dimensional stability of γ-treated U; neutron streaming through a mockup of the KAPL top plug used in SIR; supplements to neutron cross section data; absorption cross sections of U^{235} and U^{238} for thermal neutrons. (B.J.H.)

7710 BNL-249
Brookhaven National Lab., Upton, N. Y.
QUARTERLY PROGRESS REPORT FOR APRIL 1–AUGUST 15, 1953. (CLASSIFIED SECTION). Clarke

Williams, comp. Decl. Feb. 14, 1957. 37p. $0.40(OTS).

The following studies are reported: equipment and methods for study of distribution of rare-earth metals and U between molten Bi and fused salts; specific heat of a LiCl–KCl eutectic; vapor pressure of BiI_3 and partial pressure of BiI_3 over Bi solutions; summary of results of circulating fused salts and U–Bi solutions in stainless steel loops; efficiency of an electromagnetic pump; heat-transfer calculations for the LMFR; in-pile and experimental results on removal of U and Pa in the ThF_4 blanket process; thermal conductivity of ThF_4; waste-processing development studies; phase studies on U–Pb–Bi–Sn, Th–Pb–Bi, and Th–Bi alloys; static corrosion of Croloy 5 Si, Croloy 2¼, Croloy 5, special Cr–Mo–Fe alloys, and Si steel by U–Bi; dynamic corrosion of Croloy 2¼ and Croloy 5 Si by U–Bi to which Mg and Sr had been added; inhibiting effects of Zr additions to molten Bi in Fe crucibles; effects of radiation on BNL pile graphite; summary of in-pile creep studies on 2S Al; effects of radiation damage on the microformer used in the in-pile creep-testing apparatus; neutron resonances in U^{238}; neutron resonances in enriched Hf isotopes; absorption cross sections of U^{235} and U^{238} for thermal and cold neutrons; and geometrical dependence of neutron streaming through slabs of C steel. (L.M.T.)

7711 NAA-SR-72

North American Aviation, Inc., Downey, Calif.

DIFFUSION OF FISSION FRAGMENTS FROM URANIUM IMPREGNATED GRAPHITE. C. A. Smith and C. T. Young. May 4, 1951. Decl. Feb. 7, 1957. 39p. Contract AT-11-1-gen-8. $6.30(ph OTS); $3.00(mf OTS).

Measurements have been made to establish the rates of diffusion processes in graphite. Pore diffusion has been determined from flow rates of various gases through graphite samples. The gross diffusion of fission fragments from neutron-irradiated, uranium impregnated graphite has been measured at various temperatures up to 1760°C. In addition, the diffusion of the particular fragments, Ba^{140} and Sr^{89} were measured. Activation energies for the fission fragment diffusion were determined. The fraction of fission fragments penetrating the graphite crystals as a result of the neutron irradiation was measured as a function of the density of uranium impregnation. By leaching the graphite subsequent to irradiation, the uranium fuel particles were removed, and diffusion studies were carried out on the residue of the graphite crystals containing fission fragments. Data obtained in this way made it possible to separate the effect of diffusion from the fuel particles and from the graphite crystals as measured in the earlier gross diffusion studies. (auth)

NUCLEAR PROPERTIES

7712 CF-55-5-44

Oak Ridge National Lab., Tenn.

ON THE DEPTH OF THE NUCLEAR POTENTIAL WELL. Lawrence Dresner. May 9, 1955. Decl. Feb. 14, 1957. 19p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40 (mf OTS).

A parameter study is made to determine the possibility of fitting total neutron cross section data and other data with a single choice of V_0 and ζ, constants which appear in a complex square well representation of neutron-nucleus interactions. A complete discussion of the calculations is given. It is concluded that $V_0 = 36$ Mev and $\zeta = 0.2–0.3$ will fit inelastic scattering data, while a choice of $V_0 = 42$ Mev and $\zeta = .03$ is required to fit data on average total neutron cross sections. (B.J.H.)

7713 CF-55-5-69

Oak Ridge National Lab., Tenn.

CROSS SECTION PROGRAM AT OAK RIDGE NATIONAL LABORATORY (REPORT TO MEMBERS OF NCSAG FOR MEETING AT BERKELEY, MAY 24 AND 25, 1955). J. L. Fowler and H. S. Pomerance. May 10, 1955. Decl. Feb. 14, 1957. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A brief summary is presented of the cross section research program at ORNL. The program includes: fission cross section measurements of U^{234}, U^{236}, and U^{238}; study of nuclear reactions in Be^9 using a He^3 ion beam from a cascade accelerator; determination of $\nu(E)$ for U^{235}; study of inelastic neutron scattering by Zr^{90}; study of the $Be^9(n,\alpha)He^6$ reaction; time-of-flight analyses of neutron beams in the 1- to 50-kev energy region; measurements on the $V^{51}(p,n)$ reaction; and cross section measurements on Zr^{93} and Pt using a pile oscillator. (M.P.G.)

7714 HW-29569

Hanford Atomic Products Operation, Richland, Wash.

ANALYSIS OF ERRORS TO BE EXPECTED IN MEASURING THE NEUTRON ABSORPTION CROSS SECTION OF C^{12}. E. J. Seppi. Oct. 7, 1953. Decl. Feb. 20, 1957. 15p. Contract W-31-109-eng-52. $3.30(ph OTS); $2.40(mf OTS).

The probable error in the value of the C^{12} neutron absorption cross section was estimated to be about 27×10^{-27} cm^2 or about ± 8%. (auth)

7715 HW-44281

General Electric Co. Hanford Atomic Products Operation, Richland, Wash.

RESONANCE CAPTURE OF NEUTRONS BY THORIUM CYLINDERS. M. V. Davis. July 12, 1956. Decl. Feb. 26, 1957. 21p. Contract W-31-109-Eng-52. $0.35(OTS).

The dependence of the resonance integral for Th upon geometry and Th temperature has been determined through reactivity coefficient measurements performed in the Hanford Test Pile. The geometry dependence was investigated using Th cylinders of various diameters in the form of solid elements and shells. The temperature dependence (Doppler broadening) was studied by heating Th elements and permitting them to cool while observing the change in pile reactivity. (auth)

NUCLEAR REACTORS

7716 A-743

Columbia Univ., New York. Div. of War Research.

THE HEAVY WATER–SLURRY PILE. H. C. Urey. June 30, 1943. Decl. Feb. 14, 1957. 104p. Contract W-7405-eng-50. (2R-449). $16.80(ph OTS); $5.70(mf OTS).

A brief evaluation is made of the slurry pile including the tube construction, the slurry properties, the product removal, shielding, and construction materials. Complete data are included on the stability of uranium oxide systems, the preparation and properties of slurries, and corrosion and erosion of construction materials. (B.J.H.)

7717 AMF-GR-3-54

American Machine and Foundry Co. General Engineering Lab., Greenwich, Conn.

FOUR MONTH STATUS SUMMARY REPORT ON INDUSTRIAL PARTICIPATION PROGRAM. Nov. 8, 1954. Decl. Feb. 13, 1957. 10p. Appendix A: FUEL HANDLING DEVICES CONSIDERED FOR STANDARDIZATION. Engineering Report No. 6813. P. Calabretta. Nov. 1, 1954. 3p. Appendix B: LARGE REACTOR CONTROL STUDY. Engineering Report No. 6816. F.

Holzer. Nov. 9, 1954. 15p. Appendix C: AGENDA FOR DISCUSSIONS AT BRITISH ATOMIC ENERGY INSTALLATIONS. F. S. Holzer. Aug. 2, 1954. 3p. Appendix D: FUEL ELEMENT FABRICATION PROGRAM FOR NUCLEAR POWER GROUP. July 13, 1954. 6p. Appendix E: UNITIZED DESIGN FOR RESEARCH REACTOR COMPONENT. June 20, 1954. 4p. Appendix F: PACKAGED ATOMIC POWER. June 20, 1954. 4p. $9.30(ph OTS); $3.60(mf OTS).

A general survey of the AMF program in nuclear power production is given. Proposed study of reactor systems for the purpose of standardizing fuel handling devices is detailed. Reactor control components are analyzed for the purpose of standardization. Topics discussed at a meeting of AMF and British Atomic Energy officials are listed. Fuel fabrication for private utility reactor power plants is discussed. (D.E.B.)

7718 ANL-4274
Argonne National Lab., Lemont, Ill.
EXPERIMENTAL BREEDER PROJECT REPORT FOR THE PERIOD DECEMBER 1, 1948 THRU FEBRUARY 28, 1949. H. V. Lichtenberger. Apr. 12, 1949. Decl. Jan. 17, 1957. 34p. Contract W-31-109-eng-38. $6.30(ph OTS); $3.00 (mf OTS).

7719 ANL-4304
Argonne National Lab., Lemont, Ill.
MATERIALS TESTING REACTOR PROJECT. Quarterly Report for Period Ending June 1, 1949. J. R. Huffman. Decl. Feb. 4, 1957. 59p. Contract W-31-109-eng-38. $10.80(ph OTS); $3.90(mf OTS).

A comprehensive survey of the general specifications and requirements for the Materials Testing Reactor located at the Reactor Testing Station was completed. The staff at ANL prepared proposals for air cooling in the graphite, a thermal shield design, a still supporting structure design, a cadmium curtain design and experimental hole plug designs. These proposals are described in some detail in this report. Oak Ridge National Laboratory submitted a proposal for the design of a shielding test facility to replace one thermal column. The experiment on packing characteristics of graphite pebbles under thermal cycling was run the equivalent of two years of reactor operation, and the pebbles discharged. No noticeable jamming or bridging occurred, and the pebbles did not crack or spall materially. A functional site plan was prepared based on site information available. (B.J.H.)

7720 ANL-5009
Argonne National Lab., Lemont, Ill.
CASTING OF BLANKET BRICKS, RING, PLUG, AND CONTROL RODS FOR THE EXPERIMENTAL BREEDER REACTOR (CP-4). Robert E. Macherey. Mar. 19, 1953. Decl. Feb. 12, 1957. 47p. Contract W-31-109-eng-38. $0.45(OTS).

The cup assembly of the EBR consists of blanket bricks, inner rings, and safety plug, all of natural U. The design and manufacture of the parts are described. Thermal cycling data and manufacturing conditions are presented for the control and safety rods. (L.T.W.)

7721 ANL-5055
Argonne National Lab., Lemont, Ill.
PLASTER OF PARIS REPLICAS OF REACTOR FUEL SLUGS. George J. Deily. June 12, 1953. Decl. Feb. 5, 1957. 16p. Contract W-31-109-eng-38. $3.30(ph OTS); $2.40(mf OTS).

A method is described for preparing plaster of Paris replicas of irradiated fuel slugs. An impression mold is made by pouring the plaster about a fuel slug held in a specially designed holder. The slug is removed from the holder and the impression mold is used to cast a replica, also made of plaster of Paris. The replicas, including one end only, are accurate reproductions of the original slug for surface features down to about one-half mil and for dimensions within one or two mils. Two dental impression plasters were found to give satisfactory results. Within the limits mentioned, the process will serve the purpose of preserving models of radioactive specimens for use as "museum pieces." However, the procedures are very exacting in achieving such results. (auth)

7722 ANL-5108
Argonne National Lab., Lemont, Ill.
CRITICAL STUDIES WITH G. E. TYPE FUEL ELEMENTS. F. H. Martens and G. F. Helfrich. Aug. 1953. Decl. Feb. 16, 1957. 39p. Contract W-31-109-eng-38. $6.30(ph OTS ; $3.00(mf OTS).

Results are presented from various critical measurements which were made on the ZPR-I Reactor after loading with GE-type fuel elements. These included determination of the minimum critical mass with no voids or controls in the core, evaluation of several control rods, and a study of the effect of various thicknesses of Fe reflectors. (L.M.T.)

7723 ANL-5123
Argonne National Lab., Lemont, Ill.
THE DEVELOPMENT OF EQUIPMENT AND METHODS FOR CENTRIFUGALLY CASTING REACTOR FUEL SLUGS. A. B. Shuck. Sept. 29, 1953. Decl. Feb. 12, 1957. 61p. Contract W-31-109-eng-38. $0.50(OTS).

The design and construction of equipment and development of methods for multiple-mold, centrifugal casting of reactor fuel slugs are described. Advantages of the centrifugal casting method over the conventional fabrication methods were found to be fewer operations, fewer and more easily recovered residues, less expensive equipment, and the production of fuel slugs in shapes and in alloys not well adapted to other methods of manufacture. The method consisted of vacuum melting the alloy in stoppered crucibles and bottom pouring into a spinning rotor carrying 16 radially arranged copper molds. The castings so produced were used without further processing, except for cropping the sprue end to obtain the specified length. The method was found particularly adaptable to the production of enriched U–2 wt. % Zr alloy fuel slugs for the Experimental Breeder Reactor (EBR). The as-cast surfaces were satisfactory for use in the NaK-bonded fuel rods, and the as-cast 2 wt. % Zr alloy slugs were found to be more dimensionally stable than slugs fabricated by conventional methods. (auth)

7724 ANL-5457
Argonne National Lab., Lemont, Ill.
A PRELIMINARY NUCLEAR DESIGN STUDY OF DIPHENYL MODERATED AND COOLED REACTORS. E. F. Fricke. Nov. 1955. Decl. Feb. 13, 1957. 56p. Contract W-31-109-eng-38. $0.70(OTS).

Studies have been made on the basic nuclear characteristics of diphenyl moderated and cooled reactors for an average operating temperature of 600°F. Calculations include the determination of fast diffusion constants, Fermi Age, critical loadings, and radial and axial fast and slow flux distributions, for different types of reactors with power levels ranging from zero to 140 mw. (auth)

7725 ANL-5599
Argonne National Lab., Lemont, Ill.
THE MANUFACTURE OF ENRICHED ZPR-III FUEL PLATES. Frank L. Yaggee. Oct. 1956. Decl. Feb. 13, 1957. 47p. Contract W-31-109-eng-38. $0.45(OTS).

A procedural account is given of the fabrication of certain enriched ZPR-III fuel plates for use in the ANL fast

critical experiments at Arco, Idaho. A total of 208.92 kilograms of fully enriched, unalloyed uranium was processed. Of this amount 202.74 kilograms were received in the form of Oak Ridge type reduction buttons and 6.18 kilograms as pressed-powder plates. The completed fabrication consisted of 720 rectangular fuel plates having the nominal dimensions 3 in. × 2 in. × $\frac{1}{8}$ in. Their combined weight of 159.21 kilograms represents 76.22% of the weight of enriched material processed. The final distribution of the enriched material is tabulated. (auth)

7726 ANL-JTW-6
Argonne National Lab., Lemont, Ill.
NUCLEAR FUSE ELEMENT TRANSIENT HEATING. J. T. Weills. May 3, 1954. Decl. Feb. 12, 1957. 7p. Contract [W-31-109-eng-38]. $1.80(ph OTS); $1.80(mf OTS).

Results are presented of a transient heating test run on a nuclear fuse element consisting of three pieces of fully enriched U metal soldered together with 95% Sn and 5% Sb-soft solder. (F.S.)

7727 APDA-102
Atomic Power Development Associates, Detroit.
INFORMATION REPORT BY ATOMIC POWER DEVELOPMENT ASSOCIATES COVERING WORK FOR THE PERIOD AUGUST 1, 1954 TO JANUARY 31, 1955. Jan. 31, 1955. Decl. Jan. 7, 1957. 82p. $0.55(OTS).

Five-group diffusion theory equations were used in calculations performed to assist in selection of a fast breeder reactor engineering design, and results are tabulated. Analyses were conducted to determine whether the rapid expansion of solid fuel elements during a prompt cirtical accident can stabilize a fast reactor. Data are given on the effect of allotropic transformations on, and grain refinement of Cr–U alloys. Corrosion tests were made on Globeiron, zirconium tubing, and titanium. Dynamic and static corrosion testing was performed on various other materials, and results are given. The effect of Na on the self-welding of two metals is being investigated. The prolonged operation of a piston-type Johnson valve in Na was studied. Progress is reported on fuel element design and development. Analyses of the thermal performance of the elements was made. The fabrication of cast, plate-type, perforated-wafer, and pin-type fuel elements is described, and figures are shown of their assemblies. Testing and economic studies of the elements are being made. Blanket design and specifications are discussed. A cross sectional view of the reactor vessel design is given. Investigations are being made on fuel and blanket element support. Fuel loading and unloading equipment and reactor compartment arrangements have been further studied. The mechanical design of the control rod is discussed and a cross sectional view of the rod is given. The coolant flow characteristics through the reactor core during scram conditions are studied and typical scram histories are shown. A diagram of the liquid and steam-power systems is given. The intermediate heat exchanger is also shown. Diagrams of the once-through-type steam generator and the reactor and steam plant elevation are also shown. A facility for NaK–water reaction tests has been designed. Diagrams of the boiler test facility, the once-through-type test boiler and thermocouple locations for the proposed steam generator unit are shown. An analysis of relative merits of on- and off-site fuel reprocessing has been made. The design of the full-scale reactor and Na loop test facility is also discussed. (For preceding period see DCDE-101.) (B.J.H.)

7728 BMI-1031
Battelle Memorial Inst., Columbus, Ohio.
EDDY-CURRENT INSPECTION OF A POSSIBLE PWR FUEL ELEMENT. David E. Stutz, Richard L. Gibbs, Ronald F. Dickerson, and Samuel A. Wenk. Aug. 22, 1955. Decl. Jan. 29, 1957. 46p. Contract W-7405-eng-92. $0.40(OTS).

As part of a program to assist WAPD in developing a nondestructive test for a PWR-type fuel element, a number of eddy-current inspection procedures were studied. The fuel element was a rod consisting of a U-12 w/o Mo core clad with Zircaloy 2. End caps of Zircaloy 2 were welded to the body. The objectives of the eddy-current test were to detect both weld defects and lack of integrity of the body of the fuel element. Two eddy-current instruments were evaluated for their applicability to the inspection of the fuel element: the DuMont Cyclograph and the Förster Multitest. Tests were conducted with these instruments at frequencies of 20 kc and 100 kc. In the body of the fuel element, eddy-current testing detected gross unbonding of the cladding to the core, voids in the core, cracks and folds in the core, cracks in the cladding, and surface defects. Cracks shallower than about 9 mils in depth were not detected. At the welded interface, eddy-current inspection detected voids larger than about 25 mils in diameter, and areas of unbonding which were approximately one-third of the cross-sectional area of the cap-to-core interface. Eddy-current tests failed to detect core material extruded between the cap and cladding, small glass chips in the interface, and small voids. (auth)

7729 BMI-1119
Battelle Memorial Inst., Columbus, Ohio.
STUDY OF A LIQUID-METAL NEUTRON ABSORBER FOR THE CONTROL OF A GAS-COOLED POWER REACTOR. Benjamin B. Gordon, J. James Stone, and Harold M. Epstein. July 30, 1956. Decl. Feb. 12, 1957. 18p. Contract W-7405-eng-92. $0.30(OTS).

A graphite-moderated solid-fuel gas-cooled reactor does not have as large a negative temperature coefficient as is desirable. It has been suggested that a liquid-metal neutron absorber, forced into the core by thermal expansion, might produce the desired effect. For the reactor under consideration, an analog simulation study indicates that the absorber should produce an effective temperature coefficient of reactivity of 5×10^{-5} per deg F with a time constant of 0.5 sec or less. Two systems which might achieve these values of time constant and coefficient of reactivity were studied. The first system uses a mild steel reservoir filled with liquid cadmium. The expansion of cadmium in the reservoir with a rise in temperature forces cadmium into the reactor. In this system, the walls of the reservoir must be 0.002 in. thick and a large part of the power output of the reactor must be used to pump the coolant between the plates of the reservoir. This system is not practical. The other system studied consists of helium in a stainless steel reservoir connected to a manometer containing liquid cadmium. The reservoir of helium is heated by the coolant and the resulting thermal expansion forces the liquid cadmium in the manometer into the reactor. The walls of the reservoir in the helium-cadmium system could be constructed of 0.007-in.-thick stainless steel. Only a small fraction of the reactor power would be needed to pump the coolant past the reservoir. (auth)

7730 BNL-21
Brookhaven National Lab., Upton, N. Y.
NUCLEAR REACTOR PROGRESS REPORT. Oct. 1, 1948. Decl. Feb. 7, 1957. 134p. $21.30(ph OTS); $6.90(mf OTS).

Progress during the preceding 9-month period on construction of the Brookhaven Graphite Reactor is reported. Engineering studies on pressure drops, heat transfer characteristics, graphite oxidation rates, slug design and fabrication, charging technique, helium systems, etc., are summarized. Calculations by the Theoretical Group on revised pile constants, startup and shutdown problems, control, and

other problems are included. Meteorology studies are reported at considerable length. (G.Y.)

7731 BNL-48
Brookhaven National Lab., Upton, N. Y.
CLASSIFIED PROGRESS REPORT OF THE REACTOR SCIENCE AND ENGINEERING DEPARTMENT FOR 1949. [Apr. 1950]. Decl. Feb. 15, 1957. 107p. $16.80(ph OTS); $5.70(mf OTS).

Efforts during the report period have been divided between two fields: research, and evaluation and redesign of the reactor. Greater effort has been directed toward re-evaluation of the structural design of the reactor, tests on various components, and engineering design aimed at correcting such structural defects as have been uncovered. Along with these have been included studies for the start-up procedure and actual testing and readying of the reactor auxiliary equipment and instrumentation. Progress has been made on certain aspects of the research program, as well. (auth)

7732 BNL-52
Brookhaven National Lab., Upton, N. Y.
CLASSIFIED PROGRESS REPORT OF THE REACTOR SCIENCE AND ENGINEERING DEPARTMENT JANUARY 1–MARCH 31, 1950. Decl. Feb. 14, 1957. 43p. $0.40(OTS).

Subcritical Experiments on BNL Reactor. During the report period, it was possible to load the reactor to give an effective multiplication constant of 0.99. The incomplete evaluation of the data found to date indicates a flux distribution with a reflector saving of 62 cm, in agreement with two-group theory. The effect of the central gap for admission of cooling air appears to lie within the limits for the calculated values. The critical loading for the reactor is given. Continuous Processing Breeder Blanket. To date, the removal of Pa^{233} from irradiated ThF_4 and the sintering of ThF_4 have resulted in the following conclusions: an empirical expression for the changes of surface area with time has been found which permits extrapolation to longer times; the fluorine concentration in the gas to remove Pa^{233} is given; and Pa has been removed from the ThF_4 with a satisfactorily low hold-up time (one month or less) in several experiments. Metallurgy. In the studies of possible reactor fuels consisting of the fissile material suspended in a liquid metal, either as a solute or as particles of compounds containing the fuel, the following progress is reported: apparatus for phase rule experiments has been constructed; the investigation of the Pb–U system has proceeded far enough to clarify the discrepancy in the literature concerning the solubility of uranium in molten lead (a tentative phase diagram for this system has been constructed); and a preliminary experiment involving a Pb–U alloy in contact with Fe at 600°C for one month showed that no gross diffusion of U into the Fe occurs. A search of the literature for methods used to determine interfacial energies between solids and liquids has been initiated in order to understand more fully the problems concerning a slurry reactor. Initial experiments using UF_4, U_3O_8, UO_2 in liquid Pb, 55% Bi–Pb, and Bi have been completed. (auth)

7733 BNL-67
Brookhaven National Lab., Upton, N. Y.
CLASSIFIED PROGRESS REPORT OF THE REACTOR SCIENCE AND ENGINEERING DEPARTMENT [FOR] APRIL 1–JUNE 30, 1950. Decl. Feb. 14, 1957. 37p. $6.30(ph OTS); $3.00(mf OTS).

Experiments on the removal of Pa^{233} from ThF_4 by a volatility process were undertaken by studying the removal rate from a NiF_2 surface. Temperature coefficients of expansion of Invar and 42% Fe–Ni alloy were measured up to 250°C. The solubility of U in Pb and Pb–Bi alloy was studied to 900°C. Neutron economy in the Brookhaven Reactor is considered as a function of core dimensions, reactor materials, loading, and other criticality factors. The coefficient of friction of graphite on graphite is tabulated. (K.S.)

7734 BNL-348
Brookhaven National Lab., Upton, N. Y.
QUARTERLY PROGRESS REPORT [FOR] FEBRUARY 16–MAY 15, 1955. Decl. Feb. 23, 1957. 60p. $0.50(OTS).

Exponential measurements have been made on lattices of enriched U in water. Multigroup calculations for the LMFR breeders have been made with the AEC Univac. A survey of small liquid metal fuel reactors is in progress. The distribution of U in fused salt–liquid metal systems has been studied. The stability of $CeCl_3$ was investigated. The chemical processing of the LMFR blanket is discussed. Phase diagrams of the Bi–Th–Pb systems are presented. The Pa transfer from liquid phase in irradiated Th dispersions was studied under isothermal and thermal pulsing conditions. The corrosion testing of liquid Mg, Pb, and U–Bi alloys in Croloy loops and the effects of these corrosion products on the solubility of U in liquid Bi were investigated. An attempt was made to calculate corrosion rates from the solubility data of Cr and Fe in liquid Bi. The rate of reaction of U dissolved in liquid Bi with graphite was investigated. The effect of irradiation on the oxidation of graphite was studied; correlations were made between surface area and oxidation rate of pile-irradiated graphite. Additional neutron attenuation studies are reported. The corrosion behavior of steel in simulated Purex wastes was investigated. (C.W.H.)

7735 BNL-1901
Brookhaven National Lab., Upton, N. Y.
PROCEDURE FOR ANNEALING OPERATION. R. W. Powell. Oct. 2, 1953. Decl. Feb. 14, 1957. 11p. Contract [AT-30-2-gen-16]. $3.30(ph OTS); $2.40(mf OTS).

Information is given covering a procedure for annealing the graphite of the Brookhaven Reactor while in the reactor to alleviate operating problems caused by radiation damage of the graphite. (T.W.S.)

7736 BNL-2754
Brookhaven National Lab., Upton, N. Y.
THERMAL UTILIZATION, 0.387" DIAMETER 1.15% ENRICHED URANIUM RODS IN LIGHT WATER. H. Kouts, G. Price, and V. Walsh. Apr. 3, 1956. Decl. Feb. 12, 1957. 18p. $0.30(OTS).

The thermal utilization was measured for 1.143% enriched U rods, 0.387 in. in diameter, formed into lattices with uniform spacing and hexagonal geometry, and moderated by light water. The intracell distribution of thermal neutrons was determined by activation of very small foils of dysprosium oxide. Resulting flux data are tabulated. Results show that as the H_2O-to-U volume ratio is decreased from 4:1, the observed flux dip remains apparently constant, except for the 1:1 volume ratio lattice, implying that the neutron temperature is nearly the same for all the wider spaced lattices. (B.J.H.)

7737 BNL-2840
Brookhaven National Lab., Upton, N. Y.
THERMAL UTILIZATION, 1.3% ENRICHED, 0.600" DIAMETER URANIUM RODS IN LIGHT WATER. Herbert J. Kouts. July 10, 1956. Decl. Feb. 12, 1957. 14p. $0.30 (OTS).

Corrected data are summarized on the thermal utilization in 1.3% enriched, 0.600-in. diameter U rods in H_2O. These data replace those given in BNL-1783 and in Geneva paper P/600. Pertinent cross section data and

parameters are tabulated, and measured flux plots are included. (B.J.H.)

7738 BNL-2876

Brookhaven National Lab., Upton, N. Y.

LMFR PROGRESS LETTER FOR JUNE 1956. F. T. Miles. July 12, 1956. Decl. Feb. 13, 1957. 4p. $0.20(OTS).

The status of LMFR loops is summarized. (For preceding period see BNL-2875.) (W.L.H.)

7739 CE-277

[Chicago. Univ. Metallurgical Lab.]

PRELIMINARY PROCESS DESIGN OF POWER PLANT. T. V. Moore and M. C. Leverett. Sept. 25, 1942. Decl. Feb. 16, 1957. 68p. Contract [W-7401-eng-37]. (A-303). $10.80(ph OTS); $3.90(mf OTS).

A general description of a preliminary and tentative design for a He cooled power plant for Pu production at a maximum rate of 100 gms/day is presented. It must be emphasized that this design is by no means complete, and is not intended as a basis for detailing without considerable further development of many features. The function of this report is to indicate the amount of equipment, material and facilities required for a helium cooled plant of this general type and capacity; and to point out the nature of the problems involved in the construction and operation of a plant of this type, in order that a sound decision may be reached with regard to continuing development along these lines. (auth)

7740 CE-952

[Chicago. Univ.] Metallurgical Lab.

PRESSURE DROP IN RIBBED ANNULI. S. G. Bankoff and H. B. Sanford. Sept. 30, 1943. Decl. Feb. 16, 1957. 18p. Contract [W-7401-eng-37]. (A-1344). $3.30(ph OTS); $2.40(mf OTS).

In order to determine the pressure drop of water flowing through the tubes of a Hanford pile, measurements were made with water flowing at velocities of 8 to 35 ft/sec, through 12-ft lengths of 3- and 4- ribbed Al tubing. Straight rod, square-cornered slugs, and round-end slugs, both butted together and separated, were placed in the tube. The results were compared with data and correlations found in the literature. (auth)

7741 CE-2816

[Chicago. Univ. Metallurgical Lab.]

RADIATION VS. FILM FORMATION. Final Report. Problem Assignment No. 124-X14E. S. Kuniansky and R. B. Briggs. May 7, 1945. Decl. Feb. 19, 1957. 34p. Contract W-7405-eng-39. $6.30(ph OTS); $3.00(mf OTS).

Experiments in capillary tubes in the Clinton pile indicate that radiation in a 250,000 Kw pile will have little or no effect on the rate of film formation at pH > 7. At pH < 7 radiation appears to increase the rate of film formation if sodium dichromate is present in the water, probably due to the deposition of hydroxides of chromium. Experience with the experimental water tubes in the Clinton pile indicates that little trouble may be expected from film formation if the pH is maintained above 7 and the concentration of iron is kept below 0.05 ppm. Addition of some sodium silicate to the water reduces the rate of film formation, however, there is evidence which indicates that the presence of sodium silicate in concentrations as great as 40 ppm, calculated as SiO_2, may result in the formation of a less easily removable film. (auth)

7742 CF-47-3-8

[Oak Ridge National Lab., Tenn.]

EXPECTED FORMATION OF HEAVY ISOTOPES IN THE PROPOSED CLINTON HETEROGENEOUS PILE. H. C. Ott and S. B. English. Mar. 19, 1947. Decl. Feb. 13, 1957. 17p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40 (mf OTS).

The production of U^{236}, U^{237}, Np^{237}, Pu^{238}, and Pu^{239} in the enriched fuel assemblies of the Materials Testing Reactor Mockup is calculated for various cycles of operation. (G.Y.)

7743 CF-49-4-236

Oak Ridge National Lab., Tenn.

MATERIALS TESTING REACTOR; ESTIMATED CAPITAL AND OPERATING COSTS OF PROJECT. S. McLain. Apr. 26, 1949. Decl. Feb. 7, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

An estimate is given of the capital and operating costs of the Materials Testing Reactor, its associated facilities, the U^{235} separation plant, and the Metal Fabrication Plant. (B.J.H.)

7744 CF-49-10-184

Oak Ridge National Lab., Tenn.

MTR SHIELD SURVEY—REPORT NO. 2. G. Thornton. Oct. 19, 1949. Decl. Feb. 14, 1957. 15p. $3.30(ph OTS); $2.40(mf OTS).

Calculations and data are included for the MTR shield survey covering: gamma production and escape from the active pile section; capture gammas in the grid, guide, and Be reflector; and radiation leakage through bottom plug clearance and monitor tubes. Drawings are included of the bottom plug and monitor tubes. (T.W.S.)

7745 CF-49-10-186

Oak Ridge National Lab., Tenn.

ACTIVITY IN PROCESS WATER FROM EXPOSURE OF CORE MATERIAL. J. A. Lane. Oct. 21, 1949. Decl. Feb. 14, 1957. 5p. Contract [W-7405-eng-26]. $1.80 (ph OTS); $1.80(mf OTS).

The detectable magnitude of exposed fuel element core resulting from pits and blisters and the proper cooling water decay time for maximum sensitivity of the detection apparatus are calculated from data on activity resulting from fission product recoils and corrosion of the faulty fuel assemblies. (G.Y.)

7746 CF-49-10-209

Oak Ridge National Lab., Tenn.

FEASIBILITY REPORT ON PRODUCTION OF FUEL RODS FOR MTR CRITICAL EXPERIMENT. C. D. Smith, J. T. Howe, and F. W. Drosten. Oct. 31, 1949. Decl. Feb. 14, 1957. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

7747 CF-50-6-102

Oak Ridge National Lab., Tenn.

WATER FLOW IN MATERIALS TESTING REACTOR. Stanley Kasper. June 15, 1950. Decl. Feb. 14, 1957. 68p. $0.50(OTS).

A determination was made of the amounts of water flow and pressures at all points in the MTR. (B.J.H.)

7748 CF-50-7-86

[Oak Ridge National Lab., Tenn.]

A COMPARISON OF DESIGN NEUTRON FLUXES AND HEAT PRODUCTION. R. B. Briggs. July 19, 1950. Decl. Feb. 13, 1957. 20p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Neutron flux measurements were made on the Materials Testing Reactor Mockup and were extrapolated to MTR operation. It was concluded that heat production in the graphite reflector of the MTR was overestimated by a factor of 2 to 4 in critical regions, that heat production rates in the thermal shields will be less than the design values in all locations, and that operation of the reactor at

60,000 KW with open beam holes could result in overheating of some parts of the thermal shield. (B.J.H.)

7749 CF-50-7-160
Oak Ridge National Lab., Tenn.
ORNL SLUG PROBLEM—CAUSES AND PREVENTION. (thesis). Terminal Report for Period September 1948 to July 1950. R. O. Williams. Decl. Feb. 14, 1957. 90p. Contract W-7405-eng-26. $13.80(ph OTS); $4.80(mf OTS).

Submitted to the University of Tennessee, Knoxville.

Efforts were made to obtain quantitative data of the various factors which seemed important in connection with the failures of the slugs in the ORNL pile. The investigation included both radiation and non-radiation studies on standard and modified Clinton slugs and diffusion work including rates with liquid phase present and the effect of a third element. This diffusion work was supplemented by some structure studies in U–Al alloys with and without a third element present. Some physical measurements were included on the compound UAl_4, the intermetallic compound about which the least was known. The diffusion work has shown that the slugs do not, in general, fail in the minimum time necessary for the penetration of the can; but no quantitative data were obtained on the frequency distribution after this time. Temperature was the only factor studied extensively. It is very difficult to predict accurately the minimum time for failure at 250°C since there are several factors which are not known. It is, however, estimated that 70% of the 72 pile failures could have been caused by diffusion. The nature of the weakness of the weld is suggested. It is believed that about 40% of the slug failures could have been due to slugs which had holes when charged. Experience has shown that these holes were most likely at the welds. Further work has been done with slugs which have the can metallurgically bonded to the uranium slug by an Al–Si eutectic as are the Hanford slugs. While tests have not been too extensive, no trouble has been experienced except that one slug failed at 550°C after 270 days as compared to a minimum life of one day for a regular slug. Evidence is presented to show the superiority of this type of slug with respect to the weld. All data indicate that the use of bonded slugs is a satisfactory solution to the problem and would permit raising the pile-operating temperature. Results are presented to show that other elements besides Si are effective in reducing the diffusion rate between the U and the Al. The most effective of these is Bi. Semi-quantitative data are included to show that the main compound formed on diffusion is UAl_3. (auth)

7750 CF-50-9-93
Oak Ridge National Lab., Tenn.
STUDY OF STEAM POWER CYCLE AND EQUIPMENT FOR THE PILOT MODEL HOMOGENEOUS REACTOR EXPERIMENT AND PRELIMINARY STUDY OF STEAM POWER CYCLE FOR A FULL SCALE HOMOGENEOUS REACTOR. Roy C. Robertson. Sept. 13, 1950. Decl. Feb. 14, 1957. 86p. $0.60(OTS).

The steam cycle and associated equipment for the pilot model HRE are described and component specifications given. A preliminary study of the steam cycle of a full-scale homogeneous reactor is included along with a brief summary of steam power station costs and trends. (T.R.H.)

7751 CF-50-10-133
[Oak Ridge National Lab., Tenn.]
GAS GENERATION IN THE HRE PRESSURIZER SYSTEM. J. O. Bradfute. Oct. 25 1950. Decl. Feb. 14, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

An outline is given of the calculations which were made in order to determine the quantity of gas generated in the HRE pressurizer system during normal operation. It is concluded that such gas generation will be negligible. (B.J.H.)

7752 CF-51-7-3
Oak Ridge National Lab., Tenn.
EFFECTS OF POSSIBLE CHEMICAL EXPLOSION IN THE CORE TANK OF THE HRE. P. C. Zmola and T. A. Welton. July 3, 1951. Decl. Feb. 20, 1957. 4p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

The effects of chemical explosions in the HRE core tank are estimated for an instantaneous detonation and for a radially expanding, cylindrical shock front. The methods used are outlined briefly, and results are tabulated. It was concluded that an explosion in the gas core would not be of sufficient violence to rupture or deform the core tank. (B.J.H.)

7753 CF-51-7-103
Oak Ridge National Lab., Tenn.
GAS EVOLUTION IN HOMOGENEOUS REACTORS ABOVE 1000 PSI. I. Spiewak and J. O. Bradfute. July 21, 1951. Decl. Feb. 14, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

In order to evaluate the advantages of a straight-through flow of liquid in homogeneous reactors, calculations were made of gas holdup in straight-through reactors operating at a variety of pressures and liquid throughputs. Data are included on gas holdup as a function of pump inlet pressure at various temperature rises through the reactor core. (B.J.H.)

7754 CF-51-7-105
Oak Ridge National Lab., Tenn.
EFFECTS OF A POSSIBLE CHEMICAL EXPLOSION IN THE REFLECTOR TANK OF THE HRE. R. E. Aven and P. C. Zmola. July 23, 1951. Decl. Feb. 16, 1957. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Possible chemical explosions in the reflector tank of the HRE are discussed. Maximum hydrogen – oxygen explosion pressures and resulting stresses in the reactor components are given for a range of reflector temperature. Results indicate that the reflector tank and cover would not fail in this type of explosion. The rate of diluent addition required to keep combustibles at non-explosive concentrations for lower reflector temperatures is also tabulated. (B.J.H.)

7755 CF-51-8-45
Oak Ridge National Lab., Tenn.
BOILING WATER EXPERIMENTS RELATIVE [TO] BOILING REACTORS. Harold C. Urey. Aug. 8, 1951. Decl. Feb. 18, 1957. 3p. Contract [W-7405-eng-26]. $1.80 (ph OTS); $1.80(mf OTS).

It is pointed out that boiling at higher temperatures and pressures should be more favorable from the standpoint of bubble disengagement per unit of power per volume of liquid. (J.E.D.)

7756 CF-51-8-213
Oak Ridge National Lab., Tenn.
REACTOR DESIGN AND FEASIBILITY PROBLEM; U^{233} POWER BREEDER. J. Bick, J. M. LaRue, P. N. Haubenreich, D. Rauch, T. Williams, and G. Putnam. Aug. 6, 1951. Decl. Feb. 14, 1957. 150p. $0.85(OTS).

7757 CF-51-8-232
[Oak Ridge National Lab., Tenn.]
USE OF BAFFLES IN THE HRE CORE TANK. I. Spiewak and J. O. Bradfute. Aug. 29, 1951. Decl. Feb. 11, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80 (mf OTS).

Examination of the flow pattern in the plastic mockup of the HRE core tank has shown the existence of a relatively

stagnant region whose center line is common with that of the vortex. Several types of baffles were used in order to distort the natural flow pattern so as to eliminate stagnation. Details are given of the system of three tapered baffle rods which gave the best results. (B.J.H.)

7758 CF-51-8-290
Oak Ridge National Lab., Tenn.
AN ESTIMATE OF THE PHOTONEUTRON FLUX IN THE WATER SURROUNDING THE BULK SHIELDING FACILITY REACTOR. J. M. LaRue, G. P. Letz, T. J. Morley, and J. N. Renaker. Aug. 24, 1951. Decl. Feb. 14, 1957. 16p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Calculations are made in an attempt to account for the anomalous behavior of the thermal neutron flux in the water surrounding the Bulk Shielding Facility. Results of the calculations are tabulated and graphed. It is shown that even for simple calculations the calculated flux is in good agreement with the experimental measurements. (D.E.B.)

7759 CF-51-10-196
Oak Ridge National Lab., Tenn.
USE OF LIGHT WATER REFLECTOR FOR THE HRE. T. A. Welton. Oct. 31, 1951. Decl. Feb. 14, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

There is no real objection to the use of a light water reflector instead of a D_2O reflector, from the physics viewpoint. Handling convenience and soup chemistry must be considered in order to make a decision as to whether exclusive H_2O operation should be planned. (auth)

7760 CF-51-11-28
Oak Ridge National Lab., Tenn.
A POWER CYCLE STUDY. P. N. Haubenreich. Nov. 5, 1951. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A flash chamber-recombiner-heat exchanger system is proposed to prevent the formation of gas bubbles in the core of a homogeneous reactor. The case of a H_2O-moderated reactor is considered in greater detail. Performance characteristics of such a system are given in graphical form. (B.J.H.)

7761 CF-52-1-137
Oak Ridge National Lab., Tenn.
REQUIRED COPPER CONCENTRATION FOR STEADY-STATE REACTOR OPERATION AS A FUNCTION OF TEMPERATURE, FLUX, AND PRESSURE. C. H. Secoy. Jan. 23, 1952. Decl. Feb. 14, 1957. 7p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

7762 CF-52-3-36
Oak Ridge National Lab., Tenn.
CONTROL RODS FOR H.R.E. Design Report. A. Zulliger. Mar. 10, 1952. Decl. Feb. 14, 1957. 32p. Contract [W-7405-eng-26]. $0.40(OTS).

Complete design details for the HRE control rods are given. Data are included on safety rod motion. (B.J.H.)

7763 CF-52-3-211
Oak Ridge National Lab., Tenn.
DESIGN OF A NON-CRITICAL DUMP TANK ASSEMBLY FOR THE BOILING HOMOGENEOUS REACTOR. J. D. Roarty. Mar. 25, 1952. Decl. Feb. 14, 1957. 5p. $0.25 (OTS).

Design specifications are given for the dump tank assembly. Data on the critical geometry for a Cd-shielded stainless steel reflected reactor are given. The variation in infinite slab thickness of reflected stainless steel reactors for various values of H_2O moderation is also given. (B.J.H.)

7764 CF-52-5-229
Oak Ridge National Lab., Tenn.
MAXIMUM HRE POWER OUTPUT AND REQUIRED CORE PRESSURE FOR OPERATION WITH 100°C SOUP. R. Van Winkle and F. C. Zapp. May 22, 1952. Decl. Feb. 13, 1957. 32p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

The limiting factor in obtaining the maximum power for operation with 100°C soup is the main steam pipe size. For a given steam generating system and condenser or discharge pressure, the maximum steam flow is obtained at the critical or "acoustic" velocity. Any engineered restrictions in the line will reduce the power below this maximum condition. If the present HRE steam system is altered slightly to provide a 4" line from the heat exchanger to the condenser, and no throttle valve is used, the maximum amount of power that can be produced at a soup temperature of 100°C is 294 Kw. Utilizing a 4" control valve (Cv = 135) for throttling, the maximum amount of power produced is 252 Kw. At this flow rate (~800 lb/hr) the valve will be operating at critical conditions thereby limiting the power to the maximum of 252 Kw. This means that one could operate the condenser at a higher pressure and still have the same power output. Reasons for not studying heat removal using forced convection of low temperature water through the tube bundle are given. An estimate of the core pressure required for the proper discharge of the decomposition gases through the let-down system is given. The required core pressure for a power level of ~250 Kw is a minimum of 340 psia. (auth)

7765 CF-52-5-235
Oak Ridge National Lab., Tenn.
ISHR GAS SEPARATOR. J. O. Bradfute. May 21, 1952. Decl. Feb. 14, 1957. 37p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

A pipe line separator design for separating the gaseous products from homogeneous reactor fuels is given. The theory of such a separator is developed. (D.E.B.)

7766 CF-52-6-10
Oak Ridge National Lab., Tenn.
FUEL ELEMENTS FOR ORNL GRAPHITE REACTOR. E. J. Boyle and C. D. Cagle. June 3, 1952. Decl. Feb. 7, 1957. 13p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Canning procedures, testing methods, and operating experience for unbonded Al-jacketed slugs and alpha canned Al–Si bonded slugs are given. The alpha-canned Al–Si bonded slug proved most satisfactory for use in the ORNL Graphite Reactor. (D.E.B.)

7767 CF-52-6-139
Oak Ridge National Lab., Tenn.
FEED PUMP SURVEY FOR HOMOGENEOUS REACTOR PROJECT. J. M. Welch and C. W. Angel. June 23, 1952. Decl. Feb. 13, 1957. 9p. Contract W-7405-eng-26. $1.80 (ph OTS); $1.80(mf OTS).

7768 CF-52-8-128
Oak Ridge National Lab., Tenn.
REACTOR PHYSICS OF TEAPOT. PART IV. LINEARIZED KINETICS. P. R. Kasten. Aug. 15, 1952. Decl. Feb. 14, 1957. 28p. Contract [W-7405-eng-26]. $0.35(OTS).

7769 CF-52-10-155
Oak Ridge National Lab., Tenn.
REACTOR PHYSICS OF TEAPOT. PART V. PROPOSED STABILITY EXPERIMENT. P. R. Kasten. Oct. 22, 1952. Decl. Feb. 16, 1957. 15p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Two methods for determining the time relationship between input nuclear power and the vapor generated as a consequence are presented. Both involve adding reactivity in a known manner and observing the resultant neutron flux variation, the difference in the two methods lying in the time variation of reactivity additions and in the manner of interpreting the data. (auth)

7770 CF-52-10-194
Oak Ridge National Lab., Tenn.
BOILING AND SLURRY PROGRESS. R. N. Lyon. Oct. 1, 1952. Decl. Feb. 16, 1957. 31p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

Slurry Pumping Studies. Data are reported on the performance of a high-pressure slurry circulating system. As a result of test runs it was concluded that maintenance of slurry concentration in a circulating system at 250°C had not been achieved and that graphitar may be an unsuitable material for long operation in the front bearing of a pump, if slurries are permitted to enter the bearing. Data are included from studies on the formation of $UO_3 \cdot H_2O$ crystals. Boiling Reactor Research. Progress is reported in studies on boiling reactors designed to provide information for a broad evaluation of boiling homogeneous reactors in comparison with other types of reactors. Topics discussed include the nuclear stability of a boiling reactor; power removal from a boiling reactor; a large-scale homogeneous boiling reactor which existed only on paper; design of boiling reactor experiments; the treatment of boiling reactor waste gases; and investigations into types of instruments necessary to perform the experiments. (C.H.)

7771 CF-52-11-161
Oak Ridge National Lab., Tenn.
DESIGN DATA FOR ISHR (SECOND EDITION). F. C. Zapp and P. N. Haubenreich. Nov. 18, 1952. Decl. Feb. 14, 1957. 13p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Design data sheets, based on an over-all gross power rating of 60 Mw, are given. These include data on the core and blanket fluids and on the main fuel and blanket loops and components. (B.J.H.)

7772 CF-52-12-117
Oak Ridge National Lab., Tenn.
PHOTONEUTRONS IN THE ISHR. R. E. Aven. Dec. 12, 1952. Decl. Feb. 16, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Equations are derived for delayed neutron contribution to reactivity of the Intermediate Scale Homogeneous Reactor. Results of calculations are presented in tables and graphs. (C.H.)

7773 CF-52-12-177
Oak Ridge National Lab., Tenn.
THE CASE FOR THE "STEADY STATE" HOMOGENEOUS REACTOR. C. H. Secoy and H. F. McDuffie. Dec. 26, 1952. Decl. Feb. 16, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The advantages and disadvantages of internal and external O_2 and H_2 recombination in homogeneous reactors are reviewed. The use of Cu as a catalyst for internal recombination is strongly recommended. (D.E.B.)

7774 CF-53-1-150
Oak Ridge National Lab., Tenn.
MTR FUEL AND CONTROL ROD ELEMENT MEETING. J. E. Cunningham. Jan. 14, 1953. Decl. Feb. 7, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The results of the design and fabrication studies performed at ORNL to improve the operational performance of the reactor are presented and discussed. From the discussions recommendations are made for design modifications of the replacement elements needed for operation during the period January 1 through June 30, 1953. (auth)

7775 CF-53-2-75
Oak Ridge National Lab., Tenn.
DECONTAMINATION OF HRE LOOP. Quarterly Report [for] Period November 1952 to January 1953. R. J. McNamee. Feb. 6, 1953. Decl. Feb. 16, 1957. 14p. Contract [W-7405-eng-26]. $0.30(OTS).

The corrosion characteristics of 347 stainless steel piping for the HRE were investigated. Specifically, several solutions and corrosion inhibitors were tested in order to develop a satisfactory procedure for removing radioactive fission product contaminants. Two solutions were found which appeared satisfactory: Scale-Loosening Solution No. 2 (1.2M HCl and 1.8M H_2SO_4), and Pickling Solution No. 15 (1.2M HCl and 0.7M H_3PO_4). Alkyl Pyridines HB was shown to be a satisfactory corrosion inhibitor for each solution. Solution No. 15, inhibited with Alkyl Pyridines HB, refluxed at 80 to 85°C, gave a corrosion rate of 33 mils/yr. (K.S.)

7776 CF-53-2-136
Oak Ridge National Lab., Tenn.
THE USE OF CORE SOLUTION AS HEAT TRANSFER AGENT IN THE BLANKET OF THE ISHR BREEDER. C. H. Barkelew. Feb. 16, 1953. Decl. Feb. 14, 1957. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

It has been suggested that equipment savings and improvements in neutron economy could be effected by separating the two regions of the ISHR by a relatively thin shell perforated in one or more spots. The core solution would then enter through one region and leave through the other. Two-group procedures were used to calculate the neutron distribution for such a system. It is concluded that if the one-fluid arrangement is to offer an advantage in neutron economy, it will be necessary to reduce the shell thickness to something like $\frac{1}{16}$ in. (B.J.H.)

7777 CF-53-3-204
Oak Ridge National Lab., Tenn.
COOLING REQUIREMENTS FOR THE ORR. R. B. Briggs. Mar. 25, 1953. Decl. Feb. 14, 1957. 4p. Contract [W-7405-eng-26]. $0.20(OTS).

Calculations indicate that a flow of 2900 gpm through the core and reflector will be sufficient to cool the ORR at an operating power of 5 Mw in a 24-element core. The presently specified 4500 gpm should permit operation at powers as high as 7.3 Mw and possibly 7.5 Mw. It is recommended that the 4500 gpm specification be retained if changes in the water system become necessary to reduce the project cost. (auth)

7778 CF-53-5-45
Oak Ridge National Lab., Tenn.
EXPERIMENTAL PROGRAM FOR HRE AT HIGH POWER PART I: CORE MIXING, GAS COEFFICIENT OF REACTIVITY, AND CONCENTRATION EFFECT IN CORE. S. Visner. May 7, 1953. Decl. Feb. 15, 1957. 9p. [Contract W-7405-eng-26]. $0.25(OTS).

Reactivity changes associated with power, such as mixing in the core, gas residence, and the concentrating of the solution in the core, are discussed. Discussions and data are presented on the heat balance by steam flow, condensate flow, cooling of the turbine condenser, gas production, and determination of the ease of starting up the reactor by concentrating the fuel. Special operational procedures are given. Start-up, operation with constant condensate hold-up

rapid power decrease, rapid decrease in fuel pressure operation at constant outlet temperature and 1000 psi, and effect of pulsafeeder flow rate are discussed. (J.A.G.)

7779 CF-53-5-202
Oak Ridge National Lab., Tenn.
XENON POISONING IN THE ISHR. P. N. Haubenreich. May 27, 1953. Decl. Feb. 14, 1957. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Xenon-135 is created in the reactor core and is distributed throughout the entire system. It is destroyed by burnup and by natural radioactive decay. Some xenon is removed in liquid withdrawn for chemical processing. More might be removed by stripping xenon from a part of the gas stream. How the Xe^{135} concentration in the fuel solution is related to the various parameters is described below. A factor of some importance and much uncertainty is the mass transfer coefficient in the gas separator. Estimates based on results with wetted-wall columns range from 30 to 200 lb mol/hr ft^2 (mol fraction). Because of this uncertainty calculations have been made for several values up to 200 lb mol/hr ft^2 (mol fraction). Graphs are given to show how the xenon poisoning is affected by the amount of liquid flowing through the high pressure storage tank. In each case it is assumed that no liquid or gas is removed for processing. The effect of xenon removal in process streams is shown. Circulation through the storage tank is 100 gpm in each of these cases. For a chemical processing cycle time of about 20 days, 140 liters/day will be withdrawn. It can be seen that the effect of xenon removal in this amount of liquid is negligible. Xenon poisoning can be cut down sharply if a part of the gas stream has the xenon removed from it. Curves are given to show the effect of removing all the xenon from 0.1, 0.5 and 1.0 CFM of gas. (auth)

7780 CF-53-6-147
Oak Ridge National Lab., Tenn.
ESTIMATED RESPONSE OF HRE TO REACTIVITY INCREASES AT LOW INITIAL POWER. V. K. Parẽ. June 18, 1953. Decl. Feb. 14, 1957. 20p. Contract [W-7405-eng-26]. $4.80(ph OTS).

7781 CF-53-6-203
Oak Ridge National Lab., Tenn.
PRESSURE DROP IN HIGH PRESSURE GAS SYSTEM FOR ISHR. R. E. Aven. June 23, 1953. Decl. Feb. 14, 1957. 13p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40 (mf OTS).

In order to specify the required blower head, the order of magnitude of the pressure drop across the entire ISHR gas system was calculated. The composition and flow rates of the gas leaving the separator, recombiner, and condenser are tabulated. The methods of calculation are outlined, and it was concluded that the best arrangement consists of a 6-in. schedule 80 pipe between the gas separator and the entrainment separator, and between the recombiner and condenser and of a 5-in. schedule 80 pipe in the remaining part of the system. This would give a total pressure drop of about 3.0 psi in the system. (B.J.H.)

7782 CF-53-7-136
Oak Ridge National Lab., Tenn.
LENGTH OF GAS SEPARATOR FOR ISHR. R. E. Aven. July 21, 1953. Decl. Feb. 14, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Calculations of the length of the pipe line gas separator in the ISHR are outlined. A graph of the length required for gas separation as a function of gas inlet velocity is included. (C.W.H.)

7783 CF-53-9-111
Oak Ridge National Lab., Tenn.
POWER REMOVAL FROM A BOILING SLURRY BLANKET. P. C. Zmola. Sept. 21, 1953. Decl. Feb. 14, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A possible arrangement of the boiling ThO_2 slurry blanket is shown, and the problems associated with the use of a slurry in a boiling system are briefly reviewed. Calculations of the operating mean density of the blanket are included in an appendix. (B.J.H.)

7784 CF-53-9-150
Oak Ridge National Lab., Tenn.
PHYSICAL PROPERTIES OF GAS MIXTURE IN ISHR HIGH PRESSURE GAS SYSTEM. R. E. Aven. Sept. 28, 1953. Decl. Feb. 14, 1957. 33p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

In making design calculations of components in the ISHR high pressure gas system, it is necessary to have physical property data on the D_2, O_2 and He mixture involved. The composition and physical properties of the mixture at 250 and 300°C and pressures of 765, 1000, 1750, and 2000 psia were determined. The results and the methods used are given. (D.E.B.)

7785 CF-53-11-210
Oak Ridge National Lab., Tenn.
PRELIMINARY REPORT ON THE 1953 LOS ALAMOS BOILING REACTOR EXPERIMENTS. R. N. Lyon. Nov. 30, 1953. Decl. Feb. 14, 1957. 21p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Boiling reactor experiments were designed to indicate the response to large reactivity changes. Preparations for these experiments are reviewed, and a chronological statement of experimental events is given. The results indicate that boiling reactors are capable of self-adjustment to rapid reactivity changes, but further analysis of the data is planned. (D.E.B.)

7786 CF-53-12-119
Oak Ridge National Lab., Tenn.
INFORMATION PAPER ON MTR FUEL ELEMENTS. J. E. Cunningham. Dec. 21, 1953. Decl. Feb. 7, 1957. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The reactivity loss in MTR fuel elements due to entrapped brazing flux is discussed. Design and fabrication studies to lower the Al/H_2O ratio were made. Materials and performance data on fuel element designs are given. Brazing sheet material development is discussed. Fabrication costs are summarized. (B.J.H.)

7787 CF-53-12-187
Oak Ridge National Lab., Tenn.
REMOVAL OF LIQUID FROM CIRCULATING SLURRY BY FILTRATION. L. G. Alexander. Dec. 30, 1953. Decl. Feb. 14, 1957. 15p. Contract [W-7405-eng-26]. $3.30 (ph OTS); $2.40(mf OTS).

Methods of removing water substantially free of solid particles from a circulating slurry by means of constant-rate, constant-pressure filtration were investigated. Steel, glass, and cloth filters were tested. The most promising results were obtained from tests using glass wool. However, none of the materials tested produced the desired results. The tests are reviewed, results given, and recommendations for future tests are made. (D.E.B.)

7788 CF-54-1-161
Oak Ridge National Lab., Tenn.
PHYSICAL PROPERTIES OF SOLUTIONS IN THE HRT. R. E. Aven. Jan. 25, 1954. Decl. Feb. 14, 1957. 7p. Contract [W-7405-eng-26]. $0.25(OTS).

Physical properties of D_2O, $UO_2SO_4-D_2O$, and $UO_2SO_4-H_2O$ are given. (D.E.B.)

7789 CF-54-2-126
Oak Ridge National Lab., Tenn.
ISHR HIGH PRESSURE GAS SYSTEM AT 300°C AND 2000 PSIA. R. E. Aven. Feb. 12, 1954. Decl. Feb. 14, 1957. 14p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40 (mf OTS).

It appears more desirable to operate the ISHR at 300°C and 2000 psia than at the design specifications of 250°C and 1000 psia. Design changes necessary for the accommodation of this higher temperature and pressure are calculated. (D.E.B.)

7790 CF-54-4-183
Oak Ridge National Lab., Tenn.
LINEARIZED STABILITY CRITERIA FOR HRT TYPE REACTORS. P. R. Kasten. Apr. 21, 1954. Decl. Feb. 15, 1957. 12p. Contract [W-7405-eng-26]. $0.25(OTS).

The stability criteria for homogeneous reactors, as determined from the linearized equations of motion, have been investigated and the results given in graphical form. Stability from small power oscillations is assured so long as operating conditions lie to the right or above the appropriate curve. For the HRT, the stability criteria are satisfied for all operating conditions presently proposed. However, as the power density is increased, the operating point shifts toward the region of instability. Above a power density of about 100–400 kw/liter, linearized instability may occur. Operation in the so-called unstable region may be allowable, however, since non-linear terms, neglected in the analysis here, will limit any power and pressure oscillations which result. (auth)

7791 CF-54-6-16
Oak Ridge National Lab., Tenn.
DUMP TANKS FOR HRT BLANKET. S. Visner. June 3, 1954. Decl. Feb. 14, 1957. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The criticality of blanket dump tanks associated with the HRT was investigated for the contingency that the fuel solution from the core becomes mixed with the blanket contents. The solution volume of 2000 liters is estimated as necessary, but a 24-in. diameter pipe 40 ft. long with a capacity of 3500 liters is recommended. A cadmium coating is also recommended to take care of a heavy fuel loading at an end caused by tilting of the pipe. (F.S.)

7792 CF-54-6-71
Oak Ridge National Lab., Tenn.
HR PROCESSING PROGRAM: STUDY OF LIQUID-SOLID CYCLONES—PROBLEM STATEMENT. Marvin E. Whatley. June 7, 1954. Decl. Feb. 14, 1957. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A study was made to provide adequate information for the optimum design of a liquid cyclone to be used in the K-49 reactor to remove PuO_2 and fission products from the core and blanket solutions. (auth)

7793 CF-54-6-110
Oak Ridge National Lab., Tenn.
GASEOUS ACTIVITIES IN THE HRT. P. N. Haubenreich. June 4, 1954. Decl. Feb. 14, 1957. 15p. Contract [W-7405-eng-26]. $0.30(OTS).

A description is given of the activity of irradiated HRT fuel solutions due to gaseous fission products such as Kr or Xe. (B.J.H.)

7794 CF-54-6-112
Oak Ridge National Lab., Tenn.
HRT GAS SEPARATOR. J. A. Hafford. June 14, 1954. Decl. Feb. 14, 1957. 4p. Contract [W-7405-eng-26]. $0.20(OTS).

An axial, or pipe line, gas separator is recommended for the HRT. The separator will have fixed rotation vanes and recovery vanes to lower the energy loss, kill the rotation, and possibly add stability to the vortex. Design data and the performance of a test model are reported. Titanium is recommended as the most suitable construction material, since the corrosion properties of stainless steel appear unfavorable for this application. (M.P.G.)

7795 CF-54-6-161
Oak Ridge National Lab., Tenn.
GAS SEPARATOR FOR HRT. R. E. Aven. June 18, 1954. Decl. Feb. 14, 1957. 6p. Contract [W-7405-eng-26]. $0.25(OTS).

A device to remove gas from the fuel solution leaving the core of the Homogeneous Reactor Test is described. The separator is placed in the main circulating loop immediately following the reactor core and consists of fixed vanes to cause rotational flow and straightening vanes to recover some of the rotational energy of the fuel solution after the gas is separated. Separator specifications are presented, and the length of the gas vortex necessary to remove the gas has been calculated as a function of inlet tangential velocity. (M.P.G.)

7796 CF-54-9-175
Oak Ridge National Lab., Tenn.
HRT CONTROL CIRCUIT PROPOSAL. R. L. Moore. Sept. 28, 1954. Decl. Feb. 14, 1957. 45p. Contract [W-7405-eng-26]. $7.80(ph OTS); $3.30(mf OTS).

Complete descriptions are given of the proposed HRT control circuit for the case of a D_2O blanket. Operational procedures are also outlined. Data are included on the pressures during an uncontrolled dump of the HRT core and blanket systems and on the temperatures of fuel in the HRT core. (B.J.H.)

7797 CF-54-9-226
Oak Ridge National Lab., Tenn.
HRT SAFETY CONSIDERATIONS. Paul R. Kasten. Sept. 30, 1954. Decl. Feb. 15, 1957. 23p. Contract [W-7405-eng-26]. $0.35(OTS).

Recommendations are made for enhancing the safety of the Homogeneous Reactor Test. Design and operational conditions affecting safety, conceivable HRT reactivity rate additions, and the possibility of a startup accident resulting from criticality being achieved before the core is filled are discussed. (C.H.)

7798 CF-54-11-46
Oak Ridge National Lab., Tenn.
EFFECTS OF RADIOLYTIC GAS REMOVAL ON THE XENON POISONING IN THE TBR CORE, TBR DESIGN MEMO NO. 6. P. N. Haubenreich. Nov. 10, 1954. Decl. Feb. 15, 1957. 17p. Contract [W-7405-eng-26]. $0.30(OTS).

Xe^{135}, with its 3×10^6 b cross section and high fission yield, is a potentially serious poison in high-flux reactors. In the TBR, which has a flux averaged over the entire core system of about 3×10^{14}, the ratio of absorptions in Xe^{135} to fissions can be as high as 5.7%. This is equivalent to a loss of $102,000 a year in additional fuel inventory charges and decreased U^{233} production as compared to a reactor free of Xe. Fortunately the Xe^{135} concentration can be significantly reduced by allowing part of the dissociated deuterium and oxygen to form bubbles, thus stripping Xe from the solution. The magnitude of the reduction depends on the rate at which bubbles form and the design of the system for handling the separated gas. For a given rate of bubble formation, a system which lets down to low pressure for recombination gives the greatest reduction in Xe^{135} concentration. This is because the stream returned to the main circulating loop

is practically free of Xe, an impossibility with high pressure recombination systems. This by itself does not mean that the low pressure recombination system is the best for the TBR, however. Other factors of cost and engineering feasibility must be considered before a final choice can be made. (auth)

7799 CF-54-12-99
Oak Ridge National Lab., Tenn.
THERMAL SHIELD FOR HRT. Phillip M. Wood. Dec. 14, 1954. Decl. Feb. 14, 1957. 8p. Contract [W-7405-eng-26]. $0.25(OTS).

Specifications for the thermal shield over the HRT are discussed briefly. Two arrangements, one with the reactor cell open at the top and the other with a shielding cover on top of the reactor cell, are described and discussed. (B.J.H.)

7800 CF-55-3-69
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR TEST CHEMICAL PROCESSING PROJECT—APPLICABILITY OF FREEZE PLUGS, PROBLEM STATEMENT. P. E. Novak. Mar. 7, 1955. Decl. Feb. 14, 1957. 7p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

A program is suggested for testing the application and limitations of freeze plugs to be used to insure positive closure of important valves in the HRT. (F.S.)

7801 CF-55-4-19
Oak Ridge National Lab., Tenn.
COOLING OF HRT FUEL SOLUTION AS A RESULT OF CHANGING HEAT EXCHANGER OPERATING CONDITIONS. Paul R. Kasten and R. S. Stone. Apr. 1, 1955. Decl. Feb. 14, 1957. 30p. Contract [W-7405-eng-26]. $0.35(OTS).

An HRT nuclear safety study has been made concerning the effects of changing the water temperature on the shell side of the core heat exchanger. The rate at which the temperature of the fuel solution leaving the heat exchanger varies with time was found to be approximately that of the shell side temperature rate under a wide variety of operating conditions. Assuming the heat exchanger water temperature is in equilibrium with the existing pressure, a limit on rates of pressure decrease can be specified in accordance with allowable rates of reactivity addition. For a reactivity addition of 0.005 Δk_e/sec, the allowable steam pressure decrease should not exceed 20 psi/sec at all operating pressures when the 400A pump is in normal operation. If the average rate of pressure decrease is greater than 20 psi/sec, the 400A pump should be stopped. A time constant of one sec is adequate for the instrumental device stopping the pump. (auth)

7802 CF-55-5-115
Oak Ridge National Lab., Tenn.
EFFECT OF SLURRY SETTLING ON REACTIVITY IN A ONE-REGION THORIUM BREEDER REACTOR. H. C. Claiborne and T. B. Fowler. May 18, 1955. Decl. Feb. 14, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The effect of uniform slurry settling on reactivity in one-region thorium breeder reactors of economic interest was investigated. The fuel considered was a mixed slurry of UO_2 and ThO_2 suspended in D_2O. It was found that when settling occurred, thermal reactors of possible economic interest (10 to 12 ft diameter containing 200 to 300 g Th/liter) became subcritical. (auth)

7803 CF-55-7-54
Oak Ridge National Lab., Tenn.
CONTRIBUTION OF BETA PARTICLES TO HEATING OF HRT CORE TANK. P. N. Haubenreich. July 15, 1955. Decl. Feb. 14, 1957. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

When the HRT is operating at a core power of 5 megawatts, the energy of the beta particles emitted in the fuel solution will build up to 0.4 Kw/liter. Beta particles striking the inside surface of the core tank cause heat to be generated at a rate of 0.05 watts per cm^2 of surface area. This heat production is confined to a layer of metal less than 0.2 cm thick. Operation with a 355 gm U/Kg D_2O blanket results in beta heating of the outside surface of the core tank at a rate of 0.006 watts/cm^2 when the core power is 5 Mw. (auth)

7804 CF-55-7-78
Oak Ridge National Lab., Tenn.
HRT AND TBR CORE TANK TEMPERATURES. P. N. Haubenreich. July 18, 1955. Decl. Feb. 14, 1957. 14p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

It has been determined experimentally that fission products in a fuel solution tend to be deposited on a surface which is at a higher temperature than the solution. This situation exists in the core of a reactor, where the walls are heated by radiation. In order to estimate the importance of this phenomenon in the HRT, calculations were made to determine the operating temperature of the core walls. The temperature distribution in the TBR core wall has also been estimated. The estimated temperatures of the fluid in the HRT with D_2O reflector, operating at 5 Mw, are shown. Under these conditions, the maximum temperature difference between the wall and the fuel solution occurs near the bottom of the tank, where the wall is about 18°F above the core fluid. Temperature distributions through the wall are shown for operation at both 5 and 10 Mw. Heat is produced in the tank wall by gamma radiation, which generates heat throughout the metal, and beta particles, which produce heat in a thin layer at the surface. At 5 Mw the gamma heat production is 1.8 watts/cm^3, and the source on the inside surface due to the beta radiation is 0.05 watts/cm^2. At 10 Mw the heat production is twice as great. In the HRT with a UO_2SO_4 blanket, the heat production in the tank walls will be about 50% greater than when the reactor is operating with a D_2O reflector at the same core power. If the temperatures of the blanket and core fluids are equal at the equator of the core tank, the wall surface will be 13°F above the fluid when the core power is 5 Mw. The heat production in the TBR core tank is an order of magnitude greater than in the HRT. The inside surface temperature is expected to be about 50°F above the core fluid temperature. On the outside, the metal surface is estimated to be 23°F above the saturation temperature at the total pressure of 2000 psia. (auth)

7805 CF-55-8-54
Oak Ridge National Lab., Tenn.
HR UO_2SO_4 BLANKET PROCESSING, UNIT OPERATIONS DEVELOPMENT STUDIES—PROGRAM STATEMENT. P. A. Haas. Aug. 10, 1955. Decl. Feb. 16, 1957. 6p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

The program for obtaining and evaluating the engineering data necessary for successful design, fabrication, and operation of the HRT UO_2SO_4 blanket processing plant for removing Pu and reactor poisons is outlined. (F.S.)

7806 CF-55-8-108
Oak Ridge National Lab., Tenn.
TEMPERATURE COEFFICIENT AND REACTOR SAFETY. Paul R. Kasten and M. W. Rosenthal. Aug. 18, 1955. Decl. Feb. 14, 1957. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The safety of one- and two-region reactors was investigated for the case of reactivity addition by cooling of the

core fluid. For the same cooling incident, the one-region reactor had the higher degree of safety as represented by a lower core pressure rise. The results indicate that increasing the size of a reactor and thereby lowering the absolute value of the temperature coefficient is desirable, from the standpoint of reactor safety, if the most dangerous situation corresponds to reactivity additions by means of the temperature coefficient itself. (auth)

7807 CF-55-9-112
Oak Ridge National Lab., Tenn.
IRRADIATION EXPERIMENT ON ETR BORON CONTAINING FUEL ELEMENTS. R. J. Beaver. Sept. 12, 1955. Decl. Feb. 7, 1957. 5p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Design details, materials data, and physical and nuclear data are given for test assemblies of ETR fuel elements incorporating the addition of boron to th U–Al fuel alloy. (B.J.H.)

7808 CF-55-9-125
Oak Ridge National Lab., Tenn.
COMMENTS ON BOILING SLURRY BLANKETS FOR HOMOGENEOUS REACTORS. P. C. Zmola. Sept. 27, 1955. Decl. Feb. 19, 1957. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The advantages and problems associated with employing boiling as a method of power removal from the ThO_2 slurry blanket of large two region homogeneous reactors is examined in the light of existing information. It is concluded that boiling is the most promising of the systems of energy removal principally because of the simplicity attained in keeping the slurry within the blanket vessel and elimination of mechanical pumping of the slurry. It does not appear desirable to adapt a boiling blanket as a modification of the present HRT design. A large fraction of the experience and information required for design of a boiling blanket will be obtained from operation of the presently planned circulating slurry blanket. (auth)

7809 CF-55-9-162
Oak Ridge National Lab., Tenn.
REVIEW OF THE HRT-CP HYDROCLONE SYSTEM. H. E. Williamson. Sept. 30, 1955. Decl. Feb. 19, 1957. 13p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

As a result of the flowsheet revision made to incorporate the advantages of induced underflow recirculation, it is necessary to review the characteristics of the HRT hydroclone system. The results of a study on the revised HRT hydroclone system are illustrated. It seems advisable to operate the hydroclone a large fraction of the retention tank cycle time (at least 90%). The best hydroclone size and underflow to feed rate ratio seem a matter of conjecture at this time. It is suggested that a good design choice is a 400 mils diameter hydroclone designed for an underflow to feed ratio of 1%, with provisions for replacement by a smaller unit. (auth)

7810 CF-56-2-63
Oak Ridge National Lab., Tenn.
CRITICAL PARAMETERS OF A PROTON MODERATED AND PROTON REFLECTED SLAB OF U-235. J. K. Fox and L. W. Gilley. Feb. 7, 1956. Decl. Feb. 14, 1957. 5p. Contract [W-7405-eng-26]. $0.25(OTS).

Slab thickness, critical solution height, H_2O height, and critical volume and mass were determined for H to U^{235} ratios of 44.7 and 51.5 to 1. (D.E.B.)

7811 CF-56-2-87
Oak Ridge National Lab., Tenn.
PREDICTED BEHAVIOR OF FISSION PRODUCTS IN THE HRT. D. E. Ferguson, R. E. Leuze, and R. A. McNees. Feb. 16, 1956. Decl. Feb. 14, 1957. 28p. Contract [W-7405-eng-26]. $0.35(OTS).

The results of a study made to predict the behavior of fission products in the HRT are presented. Specifically, the amount of rare gases escaping to the atmosphere and accumulated in the carbon beds, the rate of accumulation of the daughters of Xe and Kr in the off-gas system, and the concentration of all non-gaseous fission products in the reactor fuel solution including iodine were estimated. In addition, the predicted accumulation of insoluble fission and corrosion products in various components of the fuel system are discussed. In most cases insufficient data are available to make firm estimates, and the results reported here should be considered very tentative until such data are available. (auth)

7812 CF-56-2-105
Oak Ridge National Lab., Tenn.
RADIATION INCIDENT OF FEBRUARY 1, 1956. A Preliminary Report. Dixon Callihan. Feb. 15, 1956. Decl. Feb. 14, 1957. 7p. Contract [W-7405-eng-26]. $0.25(OTS).

A homogeneous UO_2F_2 water moderated critical assembly was put on a prompt critical period by an over-addition of fuel to the assembly. No serious exposures resulted. The equipment, chronology of events, and cause of the accident are described. Observations and results of analyses of samples of the irradiated solution are included. (C.H.)

7813 CF-56-6-107
Oak Ridge National Lab., Tenn.
EXPLORATORY STUDY: HOMOGENEOUS REACTORS AS GAMMA IRRADIATION SOURCES. E. D. Arnold and A. T. Gresky. July 5, 1956. Decl. Feb. 15, 1957. Contract [W-7405-eng-26]. $0.55(OTS).

For single-purpose reactors, those designed exclusively as irradiation sources for food sterilization, homogeneous reactors can offer no significant advantages over other types of reactors with respect to either economic or operational feasibility. Unit costs of 4.5 to 6.0 cents per pound of irradiated meat compare closely with that found for heterogeneous types. However, very significant economic advantages exist for double-purpose reactors, those which can produce useful electrical power and also serve as food irradiation sources. Such reactors, at heat power levels of about 300 MW and with by-product utilization of off-gas, hydroclone underflow, and/or core solutions, can conceivably irradiate meat at a remarkably low cost of 0.3 cents per pound, a factor of about 20 less than for single-purpose types. Such reactors also offer a wide range of flexibility in the irradiation levels which may be required of a facility. It may be pointed out that heterogeneous types cannot compete in this way, since no feasible engineering method is evident for such by-product utilization of γ radiations. (auth)

7814 CL-RWS-31
Clinton Labs., Oak Ridge, Tenn.
ISOTOPES PRODUCED IN THE REACTOR AND ABSORBED OF A THERMAL U^{233} BREEDER AND THEIR EFFECTS ON BREEDING EFFICIENCY AND PURITY OF PRODUCT. H. G. Ott. Jan. 3, 1947. Decl. Feb. 16, 1957. 23p. Contract [W-7405-eng-39]. $4.80(ph OTS); $2.70(mf OTS).

The formation of uranium isotopes in a Th–U^{233} breeder reactor is discussed with regard to neutron economy and the optimization of breeding parameters. Capture of thermal neutrons is the principal process considered. (K.S.)

7815 CP-2065
Chicago. Univ. Metallurgical Lab.
WARPING INSTABILITY IN LONG RODS. Gale Young.

Aug. 17, 1944. Decl. Feb. 15, 1957. 10p. (A-3008). $1.80(ph OTS); $1.80(mf OTS).

If a pile rod gets bowed within its cooling tube it becomes warmer on the side which approaches the tube wall, and thermal expansion tends to wrap it in the same direction as the original displacement. This was discussed roughly in N-601, and it was there concluded that the mechanism was not important for an isolated short slug. In connection with current development of continuous jacket (cartridge) assemblies, it seems desirable to look at this question again. In this case it is possible that the effect may be of some concern. (auth)

7816 CP-2882

[Chicago. Univ. Metallurgical Lab.]

THE EFFECT OF THERMAL CYCLING ON THE THERMAL TRANSFER FROM A SAND BLASTED SLUG TO AN UNBONDED JACKET. Robert R. Bentley. Apr. 2, 1945. Decl. Feb. 14, 1957. 11p. Contract W-7401-eng-37. $3.30(ph OTS); $2.40(mf OTS).

Measurements were made to determine the effect of thermal cycling on heat transfer from slug to jacket using a slug with a sand blasted surface. The value of the transfer coefficient h decreased from initial values ranging from 10 to ∞ cal/cm^2.sec.°C., to values between 0.15 and 1 cal/cm^2.sec.°C. after 1700 cycles, and to a range of 0.15 to 0.45 cal/cm^2.sec.°C. after 3200 cycles. Comparisons are made with the results found for slugs with different surfaces. (auth)

7817 CRD-R-33

California Research and Development Co., Livermore.

POWER REACTOR LOGISTICS. D. B. Willmer. Mar. 5, 1953. Decl. Feb. 14, 1957. 21p. (LWS-24753). $0.30 (OTS).

The possible growth of the nuclear power industry to the extent of 15% of the total national requirements by 1978 is visualized. This development is considered from the standpoints of total fissionable material requirements, attainable kg per Mw ratios, and breeding gains. Time requirements for attaining a completely self-supporting industry are discussed. (D.E.B.)

7818 CS-1407

Clinton Labs., Oak Ridge, Tenn.

INCREASING PILE POWER. SELECTION OF OPTIMUM LOADING FOR CLINTON PILE. Progress Report on Problem Assignment No. 107-X6E. J. A. Lane and D. S. Webster. Apr. 1, 1944. Decl. Feb. 23, 1957. 45p. (A-2826). $7.80(ph OTS); $3.30(mf OTS).

Various methods were suggested for increasing the operating power of the Clinton pile with the objectives of getting about 150 grams of product from the pile and of producing the maximum amount of product before the end of 1944. A series of calculations were made to determine the optimum pile to achieve these objectives, and to evaluate the proposed methods of increasing the operating power. Figures, tables, and sample calculations are presented to provide the basis for estimates of pile operating power and production capacity under any given pile loading. As a result of the calculations, it has been found that the optimum size pile is a centrally poisoned pile of 40 to 50 tons of metal. The optimum pile length for this range is 15 feet, or 44 slugs per channel. Since metal economy favors the smaller pile, it is recommended that the present pile be changed to a 40 ton centrally poisoned pile which has about 700 to 800 channels with 44 slugs per channel. The amount of product which will be discharged from such a pile is estimated to be 150 grams by July 5, 300 grams by October 1, and 460 grams by December 1, 1944. These figures are based on the assumption that new plugs, jackets and a ceiling baffle will be in the pile by March 15, and new fans in operation before July 1. It is also assumed that a gradual transition from the present loading will take place. (auth)

7819 DP-114

Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Augusta, Ga.

MTR TEST EQUIPMENT. W. F. Bell. Jan. 1956. Decl. Feb. 20, 1957. 39p. Contract AT(07-2)-1. $0.40(OTS).

Sample holders and monitoring equipment were designed and fabricated for use in the irradiation of experimental shapes of fuel and target materials in the beryllium reflector of the Materials Testing Reactor. The samples were held in aluminum cases designed to replace certain beryllium blocks of the reflector. Sample holders of alternate designs were made to permit irradiation of samples in the reflector at locations where the shape or position of the reflector blocks would not permit the use of the original design. (auth)

7820 HW-15279

Hanford Works, Richland, Wash.

RADIATION AND EXPLOSION HAZARDS IN THE CRITICAL EXPERIMENTS. W. J. Ozeroff. Dec. 2, 1949. Decl. Feb. 7, 1957. 9p. Contract [W-31-109-eng-52]. $1.80 (ph OTS); $1.80(mf OTS).

The danger that a reactor consisting of a cylindrical can filled with a H_2O solution of $Pu(NO_3)_4$ and immersed in a tank of water will become supercritical is considered. The diffusion equations describing the neutrons in the reactor solution are given. Equations are also derived for explosion possibilities and radiation dosages from the reactor. (B.J.H.)

7821 HW-23353

Hanford Works, Richland, Wash.

MEASUREMENT OF ROD STRENGTHS BY ANALYSIS OF SCRAM TRANSIENTS. W. A. Clark. Jan. 28, 1952. Decl. Feb. 19, 1957. 9p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Analysis of the power-level transient following the rapid insertion of a large absorber into an operating pile provides an excellent means of determining the reactivity effect of the absorber. Recent improvements in instrumentation make possible the measurement of the strength of the entire vertical safety rod system of the production units by means of this technique, and rapid calibration of the high-speed control rods in the H unit is also feasible. The details of the method are presented together with recent experimental confirmation from the Test Pile of certain features of the theory. (auth)

7822 HW-27297

Hanford Works, Richland, Wash.

HIGH AMPERAGE SLUG WELDING. Final Report [on] Production Test 313-120-M. J. W. Lingafelter. Aug. 15, 1952. Decl. Feb. 20, 1957. 22p. Contract W-31-109-Eng-52. $0.30(OTS).

High welding amperages for effecting final closures were employed on approximately 6000 four-in. canned U slugs. Good quality weld closures were produced, and it was found, in general, that faster production rates and improved weld quality can be achieved with high amperage welding techniques. The welding operators experienced no difficulties with the higher welding amperages and the different welding rates. (auth)

7823 HW-29960

[Hanford Atomic Products Operation, Richland, Wash.]

SOME COMMENTS ON THE SELECTION OF OPERATING TEMPERATURES FOR DPR. Manfred Altman. Nov. 16, 1953. Decl. Feb. 19, 1957. 12p. Contract [W-31-109-eng-52]. $3.30(ph OTS); $2.40(mf OTS)

The results of this investigation indicate that maximum electric power generation can be obtained by selecting the turbine steam temperature within a range of ±30°C from an "idealized" steam temperature. This "idealized" steam temperature is shown to depend primarily on the maximum permissible uranium temperature. The minimum available condenser temperature and the temperature rise of the coolant in the pile affect the selection of the turbine steam temperature to a lesser extent. Equations for the "idealized" turbine temperature are given. (auth)

7824 HW-30266

[Hanford Atomic Products Operation, Richland, Wash.]
STEADY STATE TEMPERATURE DISTRIBUTIONS IN HOLLOW SLUGS. P. M. Anselone, D. O. Banks, and R. Y. Dean. Dec. 31, 1953. Decl. Feb. 20, 1957. 3p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Equations are developed for the steady state temperature distribution in externally and internally cooled hollow slugs. (B.J.H.)

7825 IDO-14331

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
PROCESSING OF THE CORE FUEL USED INITIALLY IN THE EXPERIMENTAL BREEDER REACTOR (EBR-I). A. E. Erhard, C. L. Murray, R. F. Murray, B. J. Newby, K. L. Rohde, F. P. Schilling, and F. M. Warzel. Dec. 15, 1955. Decl. Feb. 23, 1957. 46p. Contract AT(10-1)-205. $0.40(OTS).

A detailed description is given of the processing of the enriched U from the first core of the EBR I, together with the cold runs preceding the active processing. The fuel and the Al shipping cans were dissolved in HNO_3 with H_2 catalyst. Only two hexone solvent extraction cycles were used on 10 Kg per day flowsheet for decontamination and the product $UO_2(NO_3)_2$ as shipped from ICPP was of exceptional radiochemical purity. (auth)

7826 IDO-16134

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
TECHNICAL QUARTERLY REPORT [FOR] THIRD QUARTER—1953. J. R. Huffman. Nov. 1, 1953. Decl. Feb. 7, 1957. 29p. Contract AT(10-1)-205. $4.80(ph OTS); $2.70(mf OTS).

Studies of the operation of the MTR continued. In this connection, new thermal neutron flux measurements in the vertical graphite holes were used to obtain a new flux-contour map, and the study of methods of computing reactivity changes was continued. Computations were made of the reactivity changes to be expected from certain core changes The values of nuclear parameters were determined for various geometries. These latter studies are also related to the general problem of economic use of fuel which again received attention and which makes use of a simple method of estimating operating cycle times for cores containing some used assemblies. Crystal spectrometer measurements of total neutron cross sections in the energy range 0.025 to 40 ev have been completed for In, Ta, and Ti. The construction of the time-of-flight spectrometer is nearing completion and initial testing will begin soon. The design of the organic fluid circulating loop was completed. In addition, studies have continued on irradiation of plastics and graphite, fission break monitors, and scintillation techniques. A calorimetric flux monitor was completed. (auth)

7827 IDO-16150

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
IMPROVEMENTS IN MTR FUEL ASSEMBLIES AND OPERATING PROCEDURES. C. F. Leyse. Feb. 1, 1954. Decl. Feb. 7, 1957. 15p. Contract AT(10-1)-205. $0.30(OTS).

The changes in fuel assemblies, shim rods, core geometry, and reactor cycle that have been made since the MTR was placed in operation are summarized. These changes have reduced the cost of fabricating fuel for the reactor from an initial value of $51.8/Mwd to about $14.5/Mwd at present. Some ways in which further improvements might be realized are summarized briefly. (auth)

7828 IDO-16156

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.
THE FAST NEUTRON FLUX SPECTRUM OF THE MTR. M. W. Holm. Mar. 17, 1954. Decl. Feb. 26, 1957. 17p. Contract AT(10-1)-205. $3.30(ph OTS); $2.40(mf OTS).

In order to determine the group nuclear constants needed for a multi-group analysis of the MTR it is necessary to know the fast neutron flux spectrum so that these constants can be appropriately averaged. Accordingly, the fast neutron spectrum at the center of the MTR active lattice has been calculated for a 3 in. × 9 in. loading with 19-plate, thin side wall, 168-gram fuel assemblies (corresponding to an $Al-H_2O$ volume ratio of 0.581). Since the flux is relatively insensitive to the amount of U present, the results are valid for 200-gram fuel assemblies provided the $Al-H_2O$ ratio remains unchanged. Fluxes obtained with the diffusion (age) theory and by assuming pure proton moderation are provided for purposes of comparison. Corresponding fast flux weighting functions are given for the three cases. (auth)

7829 KAPL-39

[Knolls Atomic Power Lab., Schenectady, N. Y.]
FURTHER DEVELOPMENTS OF MULTI-GROUP METHOD FOR INTERMEDIATE PILES. H. Hurwitz, Jr., and R. Ehrlich. Mar. 1, 1948. Decl. Feb. 27, 1957. 29p. Contract [W-31-109-Eng-52]. $4.80(ph OTS); $2.70(mf OTS).

Some investigations which have been made in the multigroup method for calculating the multiplication constant of intermediate piles are described. The changes in the calculation procedure are of two sorts: first, experiments with somewhat altered fundamental group equations; and secondly, substitution of a numerical integration procedure for the analytic method of finding the neutron density in a particular group from the neutron density in the higher energy groups. This latter alteration is advantageous when the number of groups in a problem is large or when more than two media are involved. (auth)

7830 KAPL-44

[Knolls Atomic Power Lab., Schenectady, N. Y.]
XENON TRANSIENT CHARACTERISTICS. William G. Gibbard. Apr. 1, 1948. Decl. Feb. 7, 1957. 28p. Contract [W-31-109-Eng-52]. $6.30(ph OTS); $3.00(mf OTS).

The variation of Xe concentration is shown during transient conditions to aid in predicting reactivity variations. (auth)

7831 KAPL-346

Knolls Atomic Power Lab., Schenectady, N. Y.
MULTI-GROUP AND ADJOINT CALCULATIONS FOR SAPL-5. R. Ehrlich. May 10, 1950. Decl. Feb. 26, 1957. 43p. Contract W-31-109-Eng-52. $0.40(OTS).

The results of multigroup and adjoint calculations for SAPL-5 are presented and compared with extensive experimental measurements. This reactor has 34 moles of Be and 4.3 moles of Na per mole of U^{235} in the core, a Be reflector, and a natural U blanket. In general, the predictions of theory are confirmed by the experiments. The main discrepancies appear to lie in the below Cd flux at the center

of the core and in the low energy behavior in the blanket. The over-all criticality, the core epi-Cd flux, the fast blanket fissions, the adjoint functions (KAPL-331) and the core fission distributions appear to be in good agreement. This confirms the usefulness of the multigroup method as a dependable tool for investigating intermediate reactors. The generation time of the reactor and the dollar value is calculated. An analysis of the blanket measurements is made. A new approach to the adjoint and a new method for finding a particular solution are given. (auth)

7832 KAPL-347
Knolls Atomic Power Lab., Schenectady, N. Y.
MULTIGROUP AND ADJOINT CALCULATIONS FOR SAPL-2 AND SAPL-3. R. Ehrlich. July 17, 1950. Decl. Feb. 26, 1957. 28p. Contract W-31-109-eng-52. $0.35(OTS).

The results of the multigroup calculations for SAPL-2 and the multigroup and adjoint calculations for SAPL-3 are presented and compared with experimental measurements where possible. These calculations are based on cross sections which we have subsequently revised. Consequently the results are less reliable than those for SAPL-5 (KAPL-346) based on the newer data. Making allowances for cross section revisions and inhomogeneities in the experimental reactors, the general correspondence between theory and experiment is good. SAPL-2 and SAPL-3 are Be moderated U^{235} reactors (131 mols and 46.1 mols of Be per mol of U^{235} respectively) with natural U blankets. SAPL-3 has, in addition, about 5 cm of Be reflector. The adjoint is used to interpret the central danger coefficients for B, Fe, and Ni in SAPL-3. (auth)

7833 KAPL-552
Knolls Atomic Power Lab., Schenectady, N. Y.
THE GENIE PROJECT; NUCLEAR ENGINEERING COURSE, 1948–1949. T. Trocki, S. W. Akin, J. D. Selby, D. B. Nelson, and J. F. Cage, Jr. May 12, 1949. Decl. Feb. 27, 1957. 14p. (Memo-NEC-27). $0.25(OTS).

Early engineering design considerations for reactor cooling and heat transfer systems applicable to submarines are discussed. (D.E.B.)

7834 KAPL-818
Knolls Atomic Power Lab., Schenectady, N. Y.
OPERATING RESULTS OF THE STANDBY COOLING MOCKUP. P. R. Matthews. Apr. 15, 1950. Decl. Feb. 7, 1957. 57p. Contract [W-31-109-eng-52]. (Memo-PRM-1). $9.30(ph OTS); $3.60(mf OTS).

Operation of the Standby Cooling Mockup at Sacandaga has established the feasibility of the proposed system for cooling the reactor during an emergency shutdown. The experimental data gave valuable information in determining the nature of the heat transfer and natural convection Na flow within the reactor when in the shutdown condition. Operation of the mockup showed that while the original method for calculating the temperature distribution within the reactor at a given power level was faulty, the experimental value of the core section hot spot temperature was less than the calculated value. Also the data indicates that the heat transferred by conduction alone through the matrix amounts to about 40% of the total heat transferred by convection and conduction. Increasing the amount of heat generation in the fast breeder section while holding the total power input constant causes the total temperature drop across the matrix to decrease. Tests were made to study the effect of varying the sodium free surface within the reactor and the effect of varying the air flow rates. Several transient power runs were completed and are discussed in the report. Also a method based on mockup data, for determining Standby Cooling System performance for a full scale reactor is presented. (auth)

7835 KLX-1603
Vitro Corp. of America, New York.
HOMOGENEOUS REACTOR PROCESSING. Progress Report for Period July 1–31, 1952. Job 57. Aug. 7, 1952. Decl. Feb. 27, 1957. 30p. Contract AT-(11-1)-217. $4.80 (ph OTS); $2.70(mf OTS).

7836 KLX-1613
Vitro Corp. of America, New York.
HOMOGENEOUS REACTOR PROCESSING PROGRESS REPORT [FOR] PERIOD MARCH 1–31, 1953. Job 57. Apr. 10, 1953. Decl. Feb. 27, 1957. 11p. Contract AT(11-1) 217. $3.30(ph OTS); $2.40(mf OTS).

The efficiency of calcium fluorites, powdered PbS, galena, and monazite sands was tested in the removal of neutron-absorbing fission products from fuel solutions. The minerals were tested alone and with ion-exchange resins. (J.S.R.)

7837 KLX-1616
Vitro Corp. of America, New York.
HOMOGENEOUS REACTOR PROCESSING PROGRESS REPORT FOR JUNE 1953. Job 57. July 10, 1953. Decl. Feb. 27, 1957. 15p. Contract AT(11-1)-217. $3.30(ph OTS); $2.40(mf OTS).

A study was made of the feasibility and economics of inorganic and resin ion-exchange separation of reactor operation by-products from UO_2SO_4 fuel solutions for comparison with alternate decontamination processes. Ca-containing garnet sand was found to remove Ce selectively at higher temperatures from $UO_2SO_4-Ce_2(SO_4)_3$ solution. In a batch test at 250°C, 9.77 g of garnet sand removed 88% of the Ce tracer activity. (J.E.D.)

7838 KLX-10026
Vitro Labs., West Orange, N. J.
HOMOGENEOUS REACTOR FUEL REPROCESSING. Quarterly Progress Report [for] January 1–March 31, 1956. Dept. of Chemical Research and Development. Vitro Job 1087. Harold F. Reichard, John D. McAdoo, Stanley J. Klach, Gloria K. Koenig, and Paul E. Smith. Apr. 30, 1956. Decl. Feb. 26, 1957. 25p. For Oak Ridge National Lab. Contract W-7405-eng-26, Subcontract No. 535. $0.35(OTS).

Studies were continued on the rapid removal of I_2 from reactor core solutions by scrubbing and on the dissolution of ThO_2 blanket slurry solids by $HF-HNO_3$ solutions prior to Thorex processing. (For preceding period see KLX-10020.) (C.W.H.)

7839 LWS-22534
California Research and Development Co., Livermore, Calif.
RECOMMENDED PBR CONCEPTUAL DESIGN PROGRAM. J. E. Mahlmeister, W. H. Harker, H. L. Hollister, J. B. Stichka, and E. D. Kane. Mar. 10, 1953. Decl. Feb. 15, 1957. 23p. (CRD-A27-43) $4.80(ph OTS); $2.70(mf OTS).

A study of the engineering limitations of the core design of the Argonne Power Breeder Reactor showed the metallurgical limitations of high temperature materials, corrosive effects of Pu, and thermal stresses to be the most severe. Preliminary specifications for solid fuel and liquid fuel reactors are included. (F.S.)

7840 LWS-24612
California Research and Development Co., Livermore, Calif.
FAST REACTOR POISONING BY FISSION PRODUCTS. C. C. Old. Oct. 24, 1952. Decl. Feb. 26, 1957. 8p. (CRD-R-1). $3.30(ph OTS); $2.40(mf OTS).

Estimates are made of fast reactor poisoning by fission products. Data are included on the neutron absorption and production of fission products in reactors. (B.J.H.)

7841 M-3956
[Clinton Labs., Oak Ridge, Tenn.]
PRESSURE DROPS AND LIFTING EFFECTS OF STACKED FUEL RODS IN A SIMULATED PILE FUEL CHANNEL. Testing Report No. 1. Leonard W. Fromm. Feb. 11, 1947. Decl. Feb. 7, 1957. 12p. $3.30(ph OTS); $2.40 (mf OTS).

Graphite rods in a simulated fuel channel were investigated to show pressure drops and lifting effects. Correlations were made to conditions in the proposed Daniels Power Reactor. (F.S.)

7842 NAA-SR-230
North American Aviation, Inc., Downey, Calif.
REACTOR PHYSICS QUARTERLY PROGRESS REPORT [NO. 1]. R. A. Laubenstein, ed. Mar. 16, 1953. Decl. Feb. 7, 1957. 44p. Contract AT-11-1-GEN-8. $7.80(ph OTS); $3.30(mf OTS).

Experimental measurements in the exponential project include: transport mean free path of thermal neutrons in D_2O by a B poisoning technique; buckling of U–D_2O square lattices; half life of In^{116}(54.69 ± 0.10 min); and neutron temperature increase caused by selective diffusion into a 3-in.-diameter U sphere. Theoretical calculations made for the exponential project were: analysis of U–D_2O lattices; epithermal fission and absorption in non-reflected U–D_2O lattices; neutron flux calculations for natural U spheres in an infinite D_2O medium; and some solutions for the Boltzmann transport equation. General reactor physics studies include: kinetics of a mechanically heterogeneous, H_2O-moderated, and H_2O-reflected reactor; effect of fuel-particle size on the transient behavior of a homogeneous graphite reactor; effective radius of control rods; equations describing the production of fissionable material in a thermal reactor; effect of juxtaposed fuel elements upon the resonance absorption cross section; γ-escape probability for shielded slabs and cylinders; and a variable thickness control cell for shim control of reactors. Instrumentation for the Water Boiler Neutron Source is briefly discussed. (L.M.T.)

7843 NAA-SR-259
North American Aviation, Inc., Downey, Calif.
REACTOR PHYSICS QUARTERLY PROGRESS REPORT [FOR] FEBRUARY, MARCH, AND APRIL 1953. R. A. Laubenstein, ed. July 8, 1953. Decl. Feb. 11, 1957. 31p. Contract AT-11-1-GEN-8. $4.80(ph OTS); $2.70(mf OTS).

Exponential Experiment. A series of measurements of the thermal neutron bucklings of depleted U–D_2O lattices has been completed with the investigation of three additional lattices. Three lattices of slightly enriched U (0.9% U^{235}) in D_2O have also been studied. Besides the buckling measurements, intra-cell neutron density distributions and effective neutron temperature variations have also been measured. A nomograph for making resolving-time corrections has simplified the reduction of raw counting data to saturated foil activities. Data obtained from foils exposed to a neutron current were found to exhibit a linear correlation between the magnitude of the current and the ratio of activities counted on the two sides of the foil. A measurement of the thermal neutron diffusion length in D_2O can be used to determine the macroscopic absorption cross section of the water, and hence the light-water impurity. A discussion of the accuracy which can be obtained with this method of impurity determination is given. Water Boiler Neutron Source. Use of the water boiler neutron source for the exponential experiments and for miscellaneous irradiations has continued to increase. Preparations have been made for doing accurate danger coefficient measurements with the water boiler. These include the installation of an accurate indicator of control rod position and a thermocouple to indicate reactor temperature. Standard samples of B have been prepared. General Reactor Program. An investigation has been made into the possibility of securing intrinsic safety for heterogeneous power reactors. A possible method is to include in the reactor a relatively small homogeneous region containing a moderator, such as graphite, impregnated with U^{235} and a substance whose thermal neutron cross section increases with increasing neutron temperature. Er is a suitable material for this purpose. Formulas and curves relating to the heat production by capture γ radiation in slabs have been worked out as an aid to reactor design work. (auth)

7844 NAA-SR-289
North American Aviation, Inc., Downey, Calif.
URANIUM PRODUCTION REACTOR (UPR) QUARTERLY PROGRESS REPORT [FOR] MAY–JULY 1953. R. A. Laubenstein, W. J. Houghton, and D. H. Martin. Mar. 15, 1954. Decl. Mar. 4, 1957. 14p. Contract AT-11-1-GEN-8. $3.30(ph OTS); $2.40(mf OTS).

Measurements of the intra-cell neutron flux distributions for a proposed Uranium Production Reactor were made using a mock-up of a portion of the reactor core. Thermal neutron and Th resonance neutron flux distributions were investigated. As a result of the experimental measurements on the first mock-up, a decrease in Th content appeared necessary in the reactor design studies. Experiments are now in progress on a second mock-up in which this change has been made. (auth)

7845 NAA-SR-854
North American Aviation, Inc., Downey, Calif.
REACTOR PHYSICS QUARTERLY PROGRESS REPORT [FOR] AUGUST–OCTOBER 1953. Apr. 15, 1954. Decl. Feb. 7, 1957. 34p. Contract AT-11-1-GEN-8. $6.30 (ph OTS); $3.00(mf OTS).

A remeasurement of one of the first of 32 "clean" U–D_2O lattices investigated has demonstrated the consistency of the buckling measurements and D_2O purity. A thorough analysis of the data obtained on depleted, natural, and enriched U lattices has been made. Consideration of the possible sources of discrepancies between theory and experiment has led to a suspicion of the calculated thermal neutron diffusion lengths. A series of diffusion length measurements in nonmultiplying lattices of Pb–Cd alloy rods has been initiated. An analysis of some early exponential experiments on lattices proposed for a neutron production reactor has been carried out in order to determine whether experimental results on these more complicated structures are consistent with the analysis carried out for the "clean" lattices. The possibility of obtaining useful information from an exponential experiment, in which only a single fuel rod is used, has been considered. A simple numerical method was presented for the analysis of exponential data directly in terms of sinh functions. The use of the WBNS as a facility for measuring neutron absorption cross sections by the danger coefficient technique has been studied further. A calibration of a portion of the coarse control rod has been made in terms of the total absorption cross section of materials placed in the central exposure facility. Danger coefficient measurements on metal slugs of depleted, natural, and slightly enriched U indicated that the method could be used as a very sensitive indication of the U enrichment. An automatic control system has been installed on the fine control rod of the WBNS to automatically maintain the reactor at a constant preset power level. A different approach to heterogeneous lattice theory has been applied to slab lattices. An interesting feature of the calculations was the anisotropic effect in these lattices. (auth)

7846 NAA-SR-1049

North American Aviation, Inc., Downey, Calif.

SODIUM GRAPHITE REACTOR QUARTERLY PROGRESS REPORT FOR MARCH–JUNE 1954. Sidney Siegel and Guy M. Inman, eds. Sept. 1, 1954. Decl. Feb. 27, 1957. 108p. Contract AT-11-1-GEN-8. $0.65(OTS).

Some preliminary calculations have been made on the reduction of fuel feed costs in a thermal power reactor by a Pu feedback from irradiated fuel, thus eliminating the need for U^{235} enrichment by a diffusion plant process. A procedure was developed and an apparatus designed for measuring the temperature effect on the resonance integral of U^{238}. Intracell flux measurements have been made in a seven-rod cluster of fuel rods moderated by D_2O. A multigroup diffusion theory in slab geometry for non-miltiplying regions was formulated by matrix methods for easy calculation with a desk calculator, and the results for an Fe shield will be compared with experiment. A number of powder-compacted specimens of U alloys of Bi, Nb, Cr, Si, Al, and Mo were thermal cycled (500 cycles) from the α to β phase, and the resulting dimensional instabilities are discussed. An apparatus was designed for measuring the thermal conductivity of U at high temperatures, above 400°C. The original 12-ft fuel element design for the SGR has been modified to reduce the length to 6 ft for the SRE. Drawings of the reactor core arrangement are given, together with the general layout of the Zr-canned graphite moderator blocks. Continuing studies on the welding of required Zr sheet and plate are reported. Design data for the SRE liquid-Na cooling system, pumps, and heat exchanger are given. The Na flow pattern around the seven-rod fuel element was also determined. Control rod and safety control element designs are presented, and additional brief topics relating to the design of the SRE shield and fuel handling equipment are included. (K.S.)

7847 NAA-SR-1292

North American Aviation, Inc., Downey, Calif.

SODIUM GRAPHITE REACTOR. QUARTERLY PROGRESS REPORT [FOR] OCTOBER–DECEMBER 1954. SECTION A. A. B. Martin, ed.; SECTION B. Guy M. Inman, ed. May 15, 1955. Decl. Feb. 28, 1957. 133p. Contract AT(04-3)-49. $0.60(OTS).

Further reactivity calculations have been performed for the steady-state Pu feed-back technique and results are presented. The neutron economy for the Th–U^{233} twin core of a full-scale sodium–graphite power plant is shown. Exponential assemblies with different lattice arrangements proposed for use in sodium–graphite power reactors are shown, and neutron flux distributions given. Experimental neutron attenuation factors for iron, magnetite concrete, and ordinary concrete are presented. Results of thermal cycling of hollow slugs and powder metallurgy U alloys are discussed. Tests on fourteen Th–U diffusion couples are briefly discussed. Results are given for irradiation stability tests to 1-Mev electrons for 3 terphenyl isomers, p-quaterphenyl, and mixtures of these polyphenyls. The latest criticality enrichment values for the SRE are briefly discussed. Design and operational features of the SRE for containment of radioactivity during normal operation, fuel loading, and reactor incidents are described. Theoretical physics calculations for the two-group SRE lattice constants and their application to reactor control are discussed. Latest development results and specifications for SRE fuel elements, moderator and reflector cells, core structure, Na coolant system, control rod system, safety device systems, and instrumentation are presented. A leak detector capable of detecting 10^{-6} moles Na/liter N_2 at atmospheric pressure is described. Revised ring and loading face shielding specifications are briefly discussed, and thermal cycling and Na corrosion tests on thermal shielding material (cast iron) were performed. Twelve thermal insulating materials showed various degrees of deterioration after exposure to molten Na at 950°C for two hrs. Tests for determining and maintaining the optimum purification of the inert blanket gas (helium) are described. The fuel handling coffin development studies are summarized. (L.M.T.)

7848 NAA-SR-1457

North American Aviation, Inc., Downey, Calif.

SODIUM GRAPHITE REACTOR QUARTERLY PROGRESS REPORT [FOR] APRIL–JUNE 1955. SECTION A. A. B. Martin, ed. SECTION B. Guy M. Inman, ed. Oct. 1, 1955. Decl. Mar. 8, 1957. 113p. Contract AT(04-3)-49. $0.65(OTS).

The buildup of U isotopes in recycled Th–U^{235} fuel, and the amount of additional U^{235} necessary to restore the original reactivity for the full-scale SGR was studied. Determination of the relative activities of indium and dysprosium foils exposed in an SGR lattice allow determinations of buckling and neutron flux within the lattice. Results of these experiments are given in graphical and tabular form. General specifications for SRE experimental fuels are listed. A study was made of the behavior of Zr samples placed in the Na corrosion loop. It was suggested that tetralin might be suitable as an organic reactor coolant. Gas evolution of tetralin under MTR radiation was studied, and the corrosion of Fe and Al in tetralin while in the radiation field was determined. The status of SGR reactor construction is discussed. A metallographic analysis of the stainless steel tubing for jacketing the fuel rods is summarized. The results of tensile tests on the tubing are also given. A vacuum retort for outgassing U slugs and heat treating fuel rods was constructed. The design of the fuel rod guides and fuel elements is discussed. The design, fabrication and welding, and gas evolution of moderator and reflector cells were studied. Test apparatus is shown for the determination of galling and sliding friction between the stainless steel core bottom and steels considered for the support plate. Assembly of the core cavity liner is shown. Design of toluene coolant lines is discussed. Results of heat transfer tests on control rod rings indicate that the maximum temperature should not be more than 1400 to 1450°F for the 2.487-inch ring. Friction tests were made to determine the life time of control rod screws. Modifications of the top shield design are suggested and discussed. The radiation within and at the surface of the fuel handling coffin are calculated, and the coffin shielding is discussed. Hot cell shielding is discussed in detail. The radiation level vs. shield density is given for the primary gallery shielding. Data on the specific activity of the primary Na coolant are tabulated. Design of, and gas purification in the inert gas system are discussed. Completed fuel handling equipment is shown. Cooling characteristics of irradiated fuel elements are analyzed. (B.J.H.)

7849 NAA-SR-1626

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.

SAFETY EVALUATION OF SODIUM GRAPHITE REACTORS. R. C. Gerber. May 1, 1956. Decl. Feb. 27, 1957. 27p. Contract AT(04-3)-49. $0.35(OTS).

The safety features of the SRE and SGR are evaluated, indicating the extent of inherent safety in sodium-cooled, graphite-moderated designs. The negative temperature coefficient is treated in detail. (D.E.B.)

7850 NAA-SR-Memo-93

North American Aviation, Inc., Downey, Calif.

PROGRESS REPORT FOR JULY 1951. Sept. 15, 1951. Decl. Feb. 28, 1957. 27p. Contract AT-11-1-gen-8. $4.80(ph OTS); $2.70(mf OTS).

Nuclear and reactor performance calculations were completed for several liquid metal cooled systems using enriched U, and the results are tabulated. Reactor specifications are listed for a high temperature power plant, and static corrosion tests on possible reactor materials and coolants were completed. Reactor specifications for the Low Power Research Reactor are also discussed. Fuel element development for the LPRR is also reported. Measurements were continued on D_2O-U lattices, and some data are included. The program for the study of properties of graphite is outlined. Radiation effects in metals, alloys, insulators, and chemical systems were studied. The Materials Testing Accelerator program is also summarized. (B.J.H.)

7851 NAA-SR-Memo-225
North American Aviation, Inc., Downey, Calif.
BUCKLING OF FLAT FUEL PLATES CAUSED BY THERMAL EXPANSION, WITH APPLICATION TO THE CONVERTER REACTOR. W. C. Cooley. Feb. 14, 1952. Decl. Feb. 27, 1957. 5p. Contract [AT-11-1-GEN-8]. $1.80(ph OTS); $1.80(mf OTS).

The conditions for yielding and for buckling of flat fuel plates under thermal stresses are plotted as a function of the ratio of plate width to thickness. It is concluded that flat aluminum-clad fuel plates 0.060 in. thick and 2.62 in. wide are probably not suitable for the converter reactor unless the film drop at the end of a fuel assembly, next to the end box casting, is less than 30°F and the side plates near the mid-plane of the reactor are free to bow, letting the fuel plates expand freely. Conservative design would require that the fuel plates be curved, as for the MTR. (auth)

7852 NAA-SR-Memo-228
North American Aviation, Inc., [Downey, Calif.]
USE OF NAA FACILITIES TO OBTAIN EXPERIMENTAL DATA USEFUL TO THE CONVERTER REACTOR. R. Balent. Feb. 19, 1952. Decl. Feb. 27, 1957. 7p. Contract [AT-11-1-gen-8]. $1.80(ph OTS); $1.80(mf OTS).

Proposed studies are outlined which use the NAA facilities to obtain information on the neutron flux distribution in the converter reactor. Possible core designs are discussed briefly. (C.H.)

7853 NAA-SR-Memo-468
North American Aviation, Inc. [Downey, Calif.]
COOLING OF TUBULAR FUEL ELEMENTS. M. Heisler and T. Shimazaki. Nov. 6, 1952. Decl. Feb. 7, 1957. 14p. $3.30(ph OTS); $2.40(mf OTS).

A general definitive comparison is made of the specific power of two Na-cooled fuel cells of equal size, one containing a tubular element and the other seven clumped rods, each cell having the same volume of U. The temperature within the cell is not to be great enough to throw the U metal into the $\alpha-\beta$ transition region. Analysis of results reveal that, for a U-to-cell volume ratio of 0.45, tubular elements permit a slightly higher specific power than clumped rods. A 2-in. cell using tubular fuel elements was found theoretically to be capable of developing about 2% more specific power than the seven-rod clump. (C.H.)

7854 ORNL-50
[Oak Ridge National Lab., Tenn.]
HIGH FLUX REACTOR FUEL ASSEMBLIES VIBRATION AND WATER FLOW. Problem Assignment No. TX5-12. W. K. Stromquist and Oscar Sisman. May 1948. Decl. Feb. 11, 1957. 53p. Contract W-7405-Eng-26. $9.30 (ph OTS); $3.60(mf OTS).

Frequencies and amplitudes of vibration in the fuel assembly for the high flux reactor have been measured. The vibrations are considered to be insignificant. Pressure differences between annuli of the assembly are insufficient to cause buckling of the plates under ordinary conditions. Pressure drops through the assembly have been measured. (auth)

7855 ORNL-170
Oak Ridge National Lab., Tenn.
SLUG RUPTURES IN THE OAK RIDGE NATIONAL LABORATORY PILE. C. D. Cagle and L. B. Emlet. Oct. 13, 1948. Decl. Feb. 11, 1957. 38p. Contract W-7405-Eng-26. $6.30(ph OTS); $3.00(mf OTS).

7856 ORNL-1137
Oak Ridge National Lab., Tenn.
SOLUTION OF EXPERIMENTAL BREEDER REACTOR SLUGS. E. M. Sampson, Jr. Mar. 11, 1952. Decl. Mar. 5, 1957. 45p. Contract W-7405-eng-26. $0.40(OTS).

A full-scale, metal dissolver for Experimental Breeder Reactor fuel was designed, built, and installed for test operation. It was found that the dissolver operated satisfactorily, and feasible operating procedures were established for the dissolution of bare, or jacketed, EBR slugs. Minor modifications of the dissolver design have been required to accommodate a modified EBR slug, but it is believed that this will not significantly affect its operating characteristics. (auth)

7857 ORNL-1280
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR PROJECT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MARCH 15, 1952. W. E. Thompson, comp. July 14, 1952. Decl. Mar. 7, 1957. 248p. Contract W-7405-eng-26. $1.35(OTS).

Difficulties encountered in tests on fuel, reflector, and gas-handling systems of the HRE are explained. Calculations are made of anticipated poisoning in the HRE due to Xe^{135}. Pulse feeder pump performance is studied with the HRE mock-up. HRE design is discussed with reference to improved alternate components. A possible mechanism for the corrosion of 347 stainless steel in uranyl sulfate solutions is investigated, and the effect of various substances added to the uranyl sulfate solution is studied. Dynamic and static corrosion studies of homogeneous reactor solutions are reported. Radiation stability, controls, and instrumentation of the HRE are described. Slurry studies, boiling reactor studies, and UO_3 chemistry are reported. A general study of Pu-producing reactor is made, and predesign of an intermediate-scale homogeneous reactor (ISHR) to provide the experience and information necessary for resonable extrapolation from the HRE to a full-scale production reactor is discussed. The studies include ISHR design, engineering studies of components, solution chemistry, static corrosion tests of alternate systems, fundamental corrosion studies, metallurgy, and chemical processing. (L.T.W.)

7858 ORNL-1658
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR PROJECT. Quarterly Progress Report for Period Ending October 31, 1953. W. E. Thompson, comp. Feb. 25, 1954. Decl. Mar. 4, 1957. 150p. Contract W-7405-eng-26. $0.80(OTS).

Homogeneous Reactor Experiment. HRE reactivity experiments were conducted by rapid variation of control parameters. The shortest period observed was 0.034 sec. Additional information on the corrosion of Ni by the fuel solution was obtained. Boiling Reactor Research. Experiments have shown that O_2, formed in boiling UO_2SO_4 solution by fission recoil, will not maintain sufficient concentration to stabilize a boiling reactor, while the solution is in contact with stainless steel at 250°C. General Reactor Studies. Advanced design planning of a high-power ISHR central station plant is reviewed. Flow tests have been

made in a full scale ISHR core. The performance and design of a catalytic recombiner and entrainment separator are also presented. Extensive data on the in-pile corrosion rates of stainless steel, Zr and Zr alloys, and Ti by UO_2SO_4 are reported. The design of several loop assemblies used in the tests is described. Other aspects of the HRP project reported include the fracture behavior of Ti and Zr, the physical properties of Ti and Zr alloys, phase studies of the $UO_3-SO_3-H_2O$ system at varying temperature, the phase stability of $Th(NO_3)_4-HNO_3-H_2O$ systems, the solubility of H_2WO_4 in UO_2SO_4, the decomposition of H_2O_2 in UO_2SO_4 and $UO_2(ClO_4)_2$, and the recombination rate of UO_2SO_4 at 250°C by Cu^{++}. Investigations of the chemistry and physical properties of ThO_2 as a circulating blanket system were continued. Physical studies of UO_3 slurries have led to new methods for reducing caking, and the circulation requirements of $(UO_2)_3(PO_4)_2-UO_3$ slurries were determined. Additional investigations are reported on the chemical processing of two-region breeder reactors by CaF_2. (K.S.)

7859 ORNL-1853

Oak Ridge National Lab., Tenn.

HOMOGENEOUS REACTOR PROJECT QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JANUARY 31, 1955. H. F. McDuffie and D. C. Kelly, comps. Feb. 16, 1955. Decl. Mar, 2, 1957. 226p. Contract W-7405-eng-26. $1.35(OTS).

Homogeneous Reactor Test. A summary of a hazards report is presented. Possible blast effects from a rupture of the pressure vessel were considered sufficient to justify the inclusion of a 1.5- to 2-in. thick blast shield around the pressure vessel. A maximum pressure of 29 psig is expected in the reactor equipment cell in the case of instantaneous release of core and pressure-vessel liquids, and release of heat exchanger liquids through 6-in. steam lines. The danger of UO_4 precipitation in the cooled dump and sampler lines was investigated. If the reactor is operating at 10 Mw, the fuel solution cannot immediately be cooled below 70°C after removal from the main circulating loop. Designs for the reactor pressure vessel and core tank were completed, and stresses in the carbon steel wall of the pressure vessel were calculated. A parametric study of core pressure rise as a function of reactivity addition was conducted on the ORACLE. Parameters included the temperature coefficient of reactivity, length of piping between core and pressurizer, friction coefficient, fluid compressibility, volume of pressurizing fluid, and lifetime of prompt neutrons. The phenomenon of "walk-away," a slow reactivity rise due to fuel concentration, was investigated. Fuel concentration results when water is removed from the core at a faster rate by radiolysis than is returned after recombination. It is suggested that inherent stabilization against this effect may be obtained by consideration of the fuel feed pump rate, overpressure, etc. Design and testing of a number of pumps are reported. A Westinghouse 400-A fuel pump operated on H_2O for 80 hr and UO_2SO_4 (10 g/kg) for 70 hr when a thrust bearing failed. The replaced bearing continued for 522 hr on UO_2SO_4. Testing of wire pressure seals for reactor instrumentation resulted in the selection of Stupakoff seals for electrical wiring and crushed lava for thermocouple wires. The Jordan RAM radiation monitor was selected for process monitoring applications. Thorium Breeder Reactor. The unit cost of electrical power from a 2-region TBR is found to be insensitive to rather large variations in the nuclear constants. The results are for a 3-reactor station supplying 375 Mw at 26% net station efficiency. The cost influence of larger turbine-plant components and TBR optimization were studied. With the temperature of the core solution between 250 and 300°C, optimum saturated steam temperature is 436°F for a blanket solution between 473 and 599°F. Large changes in inventory charges and reactor depreciation had slight effect on optimum steam temperature. Corrosion. A continuing series of tests on the corrosive effects of UO_2SO_4 on stainless steels, Zircaloy, titanium alloys, Inconel, and Stellite are reported. In a group of stainless steel loops circulating 0.06M UO_2SO_4 and 0.006M H_2SO_4 at 320°C, the critical velocity of the system was greater than 100 fps, and the stainless steels showed corrosion rates of less than 1 mpy. Cavitation may have a slight effect on critical velocity, but it was also found that small amounts of hexavalent Cr in the corrodent has very pronounced critical velocity effects. Sedimentation rates of ThO_2 slurries, ranging in concentration from 250 to 1000 g of ThO_2/kg H_2O, were determined at 250 to 300°C. In-pile loop studies were made of UO_2SO_4–stainless steel corrosion in the LITR DD loop. Type 347 stainless steel coupons were exposed and examined metallographically. The data are presented in tabular form. The DD loop was sectioned and examined. Other in-pile corrosion tests of Zircaloy-2 and Ti are described. Engineering Development. The efficiency of platinized alumina catalyst in a high-pressure H_2-O_2 recombiner loop at 1000 psi was shown to be as good as that at atmospheric pressure. A low-pressure thermal convection catalytic recombiner is being designed. The addition of SO_4^{--} to ThO_2 slurries (3000 ppm) lowered the pH below 6. Concentrations above 10,000 ppm produced serious caking. Removal of the sulfate from ThO_2 is possible by calcining at temperatures above 800°C. Pressure drop and flow studies were made on ThO_2 slurries, together with attack measurements on stainless steel. A diagram of the revised slurry blanket mockup is presented. Welding of the HRT pressure vessel (carbon steel internally clad with 347 stainless steel) will be tested on full-thickness clad material, utilizing a sequence of weld deposits involving 347 weld metal in the root and groove, two layers of 308L stainless, 2 layers of ingot-iron weld metal, and finally 7016 carbon-steel electrodes. A fully austenitic, modified 347 welding wire containing Mn was successfully used, following the Heliarc welding process. Equipment has been designed for the direct loading of small diameter wire specimens in an apparatus for testing the stress-corrosion susceptibility of austenitic stainless steels. The dynamic corrosion behavior of austenitic stainless steel welds was studied. Irradiated and unirradiated specimens of Zircaloy-2 were impact tested, and it is concluded that no embrittlement of the impact specimens took place. The current status of HRT core vessel fabrication is outlined with a number of photographs. Chemical Engineering. Irradiation of a ThO_2-U^{235} blanket slurry produced a viscous slurry which was difficult to pour. Similar tests on a pure ThO_2 slurry revealed no changes from irradiation. In the former case, with 1% Cu added, complete recombination of radiolytic gases was indicated. None of the pure ThO_2 slurries exhibited sufficient catalytic activity for reactor use. With 0.09 M $CuSO_4$ and 0.0025 M H_2SO_4 added, sufficient recombination was obtained to maintain the TBR blanket at < 2000 psi total pressure. The solubility of $Nd_2(SO_4)_3$ in the fuel solution is 0.02 to 0.03 g/kg of H_2O at 285°C. No significant amount of insoluble Nd was found in the main stream. Some of the sulfate precipitated on hot metal surfaces, the remainder collected in stagnant and low-velocity regions. Tests confirmed the tendency of rare earth sulfates to precipitate on hot stainless steel and Ti surfaces, except that a small amount adhered to Zircaloy-2 surfaces in contact with sulfate-containing solutions at the same temperature during precipitation. Suspended solids (1μ) can be removed by ¼-in. hydroclones with efficiencies greater than 90%. The efficiencies of these units were studied for

various operating conditions and dimensions. A revised flowsheet of the HRT chemical processing plant provides for a single stage of hydroclones, with drying and resolution of the solids taking place in the shipping container rather than specially designed equipment. Revised data are presented on the adsorption of Pu on 347 SS, Ti, and Zr. Precipitation of PuO_2 under simulated reactor conditions was rapid enough that supersaturation of the blanket will not be greater than 0.3 mg of Pu/kg of D_2O. When Pu(VI) was added to the UO_2SO_4 solution, more than 90% was reduced to Pu(IV) in 5 min. Complete dissolution of ThO_2 can be accomplished by refluxing in 30% excess 8N HNO_3 containing 0.04N HF. Supporting Chemical Research. Yields of H_2 and N_2 from γ-ray decomposition of $Th(NO_3)_4$ solutions were determined. Eu is reversibly absorbed from HCl solutions by commercial ZrO_2, $ZrO_3(PO_4)_4$, and ZrO_2 treated with H_3PO_4. The sharpness of the elution curves resembles that of conventional ion exchangers. Studies of the $ThO_2-CrO_3-H_2O$ system at elevated temperatures indicate that too high a CrO_3/ThO_2 ratio is required for the solution to be attractive as breeder blankets. (K.S.)

7860 ORNL-2071
Oak Ridge National Lab., Tenn.
OPERATION OF THE ORNL GRAPHITE REACTOR AND THE LOW-INTENSITY TEST REACTOR, 1955; LITR FLUX TRAVERSES. A. F. Rupp and J. A. Cox [comps.]. (Compiled from reports by: W. M. Casto, C. D. Cagle, W. H. Tabor, and T. C. Weeks). Aug. 10, 1956. Decl. Mar. 2, 1957. 40p. Contract W-7405-eng-26. $0.40(OTS).

The ORNL Graphite Reactor operated very well during 1955. The downtime was low, only 8.6%. The fuel in the bonded slugs did not perform as well in 1955 as in 1954. Much of the trouble was undoubtedly due to growth of slugs which were not beta-transformed. It is known that some slugs had grown over ½ in. The automatic control system installed in 1954 continued to operate satisfactorily. The cooling system gave minor trouble when one of the 900-hp fan motors had to be replaced because of shorts in the rotor. The high radiation in the canal was the largest source of trouble. Approximately 55 tons of slugs discharged from the reactor in 1952 were sent to the Metal Recovery Plant. Enough slugs had ruptured, due to their long exposure in the canal and reactor, to contaminate badly of water. Most of the contamination was removed by the end of the year, but the radioactivity which had soaked into the canal walls was enough to give high radiation fields. A solution to this problem was being sought at the end of the year. A study is under way on the possibility of increasing the flux of the ORNL Graphite Reactor by reloading with enriched uranium fuel. Results of the study appear to be promising. The LITR operated very satisfactorily in 1955. No major troubles developed, and the usage for research increased steadily. More facilities were installed for radioisotope production, and at the end of the year a project was under way to make a smaller grid assembly, thus exposing more of the lattice for experiments. Another project was under way to provide a third vertical facility which will not have to be removed each time the top plug is taken off. The cycle time for major shutdowns was increased to three weeks, which will cause less interference with research and operations. The downtime was high because of over 1000 hr of scheduled and 150 hr of unscheduled shutdowns for research. The utility and versatility of the LITR were well demonstrated by the 149 special scheduled shutdowns for research which enabled experimenters to obtain important data. A test of remote control of the LITR from the Graphite Reactor control room was begun in October and was still under way at the end of the year. The results appear very promising to date. A large number of scrams, apparently originating in the period amplifier or period system, was one of the worst troubles encountered. However, no single cause for the spurious scrams could be found, and they continued to occur at varying frequencies throughout the year. Improvements continued in the demineralizer system, and the amount of Np^{239} in the water decreased by a factor of about 6. This appears to indicate that the Np^{239} was being formed from natural uranium contamination in the reactor and that the source has been partly removed. (auth)

7861 ORNL-2096
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR PROJECT QUARTERLY PROGRESS REPORT FOR PERIOD ENDING APRIL 30, 1956. H. F. McDuffie and W. D. Reel, comps. May 10, 1956. Decl. Mar. 2, 1957. 159p. Contract W-7405-eng-26. $1.10(OTS).

Homogeneous Reactor Test. Construction was completed and cleaning and pretreating operations are in progress. Pressurizer design and insulation, freeze jacket design, and check valve design were established. Test results on fuel and blanket pumps, heat exchangers, and pressurizers are summarized. The power-time relation for the HRT after shutdown was obtained for several situations. A schematic of the fuel processing system is given. Flow rates, pressures, temperatures, and liquid levels throughout the system are discussed. Changes made in the sampling method are described. Reactor Design and Analysis. Fuel costs for U^{235} burners using highly enriched fuel were calculated for H_2O- and D_2O-moderated reactors between 2 and 10 ft. in diameter. The fuel cost was ~4 mills/kw hr if the fuel was processed after 2000 days or more. Nuclear and economic calculations done for cylindrical, two-region, Th breeder reactors are reported. Engineering Development. Recombiner operating experience is reviewed. Analysis of the hard ThO_2 cake formed in the 200A loop revealed no impurities which might have caused its formation. Tests on Al_2O_3 in pump bearing applications produced no impressive results. Some of the fluid flow properties of ThO_2 slurries were established. Corrosion and Materials. Operating experience in the all-Ti fuel loop has reached 8500 hrs at 300°C and 1800 psig. Corrosion characteristics are excellent for concentrations up to 1.3M UO_2SO_4. Corrosion test results are given for mixtures of UO_2SO_4 and Li_2SO_4 in stainless steel loops. The addition of Li_2SO_3 appears to produce some reduction in the corrosion rate. An in-pile Zircaloy-2 experiment, with 0.5M excess H_2SO_4, was completed. Specimen data indicate the excess acid reduced radiation corrosion. Metallographs are given which confirm the belief that the hydrides in Zircaloy-2 are formed at grain boundaries. Results of impact tests on Zircaloy-2 aged at 250°C for 0, 500, 1085 and 1585 hrs are presented. No effect of aging was noted. Chemical Engineering Development. The preparation and viscosity characteristics are reported for a number of ThO_2 slurries. Effects of varying ThO_2 concentration and UO_3 and CaO additions are discussed. The solubilities of I, Se, Sb, Nb, Tc, and Re in UO_2SO_4 were determined in connection with fuel reprocessing studies. Breeder blanket studies were extended by the determination of Pu adsorption on Ti and Zircaloy-2 and Pu solubility in UO_2SO_4. Supporting Chemical Research. Vapor pressure studies on concentrated $LiNO_3$ and $Ca(NO_3)_2$ indicate these solutions may be satisfactory fuel carriers in homogeneous systems and heat transfer media in heterogeneous systems. The effective capture cross sections of U^{238} in highly irradiated MTR fuel elements were measured, and data are presented. (For preceding period see ORNL-2057.) (D.E.B.)

7862 ORNL-2165
Oak Ridge National Lab., Tenn.
SOME ECONOMIC AND NUCLEAR CHARACTERISTICS OF CYLINDRICAL THORIUM BREEDER REACTORS. D. C. Hamilton and P. R. Kasten. Oct. 11, 1956. Decl. Mar. 1, 1957. 48p. Contract W-7405-eng-26. $0.45(OTS).

Nuclear and economic calculations were made for some cylindrical, two-region, Th breeder reactors to study breeding ratios, core-wall-power density, and fuel costs as a function of reactor size for a given total power. A low power density at the core-tank wall can be achieved with core tanks of small diameter (core and blanket diameters of 4 to 6 and 10 feet, respectively) for reactor lengths of 20 feet or greater. For a total reactor power of 450 Mw, the core-wall power density would be ~20 kw/liter for a core diameter of four feet, and ~10 kw/liter for a core diameter of six feet. The corresponding breeding ratios would be about 1.15 and 1.04, respectively. However, minimum fuel costs were associated with high core-power densities; increasing the reactor length and reducing the power density increased the inventory charge until that charge controlled the fuel cost. (auth)

7863 TID-5351
Argonne National Lab., Lemont, Ill.
EBR OSCILLATIONS. J. R. Dietrich. Jan. 4, 1956. 8p. Decl. Mar. 5, 1957. 8p. Contract [W-31-109-eng-38]. $0.25(OTS).

The relationship of temperature coefficients to reactivity oscillations in the EBR is discussed. Equations are developed for describing oscillations originating in negative temperature coefficients. (D.E.B.)

7864 TID-7506(Pt. II)
Argonne National Lab., Lemont, Ill.
PAPERS PRESENTED AT THE TECHNICAL BRIEFING SESSION ON THE BOILING WATER REACTOR PROGRAM AND THE FAST REACTOR PROGRAM, HELD AT IDAHO FALLS, IDAHO, NOVEMBER 1-2, 1955. July 1956. Decl. Jan. 17, 1957. 40p. Contract W-31-109-eng-38. $0.40 (OTS).

The processes involved in the fabrication and testing of EBWR fuel elements are briefly summarized. A general discussion is also given on the processing of spent fuel from fast reactors. (B.J.H.)

7865 WIAP-9
Westinghouse Electric Corp. Industrial Atomic Power Group, Pittsburgh.
PROPOSED 80,000 KILOWATT HOMOGENEOUS REACTOR PLANT. PROCESS AND PLANT DESCRIPTION. D. H. Fax, ed. Feb. 1955. Decl. Feb. 7, 1957. 46p. $0.45(OTS).

A proposed single-region aqueous homogeneous reactor using a dilute uranyl sulfate solution as fuel and designed for a net electrical generation of about 80,000 kw is described. The principal design parameters are summarized. Complete diagrams are given for all phases of the reactor. (B.J.H.)

Refer also to abstracts 7536, 7611, 7623, and 7628.

NUCLEAR TRANSFORMATION

7866 ANL-5335
Argonne National Lab., Lemont, Ill.
THERMAL NEUTRON IRRADIATION OF NATURAL THORIUM. Frank R. Taraba. July 30, 1954. Decl. Feb. 7, 1957. 21p. Contract W-31-109-eng-38. $0.35(OTS).

A quantitative evaluation of the principal nuclear reactions occurring in the thermal neutron irradiation of natural Th resulting in the formation of U^{233} and various other nuclear products is presented. The reactions are represented mathematically by a system of linear differential equations. An equation for calculating the burnup of natural Th resulting only from the fission of U^{233} is presented. It is assumed for mathematical simplicity that the supply of Th^{232} remains constant throughout the irradiation. The production of Th^{233}, Pa^{233}, U^{233}, and U^{234} resulting from the neutron irradiation of natural Th, the radioactive decay of Pa^{233} and the resulting formation of U^{233}, and the total atom burnup and specific power of natural Th resulting from the fission of U^{233} are presented graphically. (C.H.)

PARTICLE ACCELERATORS

7867 CRD-T2B-45
California Research and Development Co., Livermore, Calif.
THERMAL SHOCK: STRESSES IN PRISMATIC TARGET ELEMENTS SUBJECTED TO RAPID HEATING. J. P. Frankel and C. C. Old. Aug. 3, 1951. Decl. Feb. 14, 1957. 9p. (LWS-23064). $1.80(ph OTS); $1.80(mf OTS).

Results indicate that the stresses set up in the tubes of the A-12 primary target because of inertial effects associated with rapid heating are probably negligible. (auth)

7868 LRL-51
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
A HIGH CURRENT RF ION ACCELERATOR FOR USE AS AN MTA INJECTOR. R. E. Hester, R. E. Giebeler, and D. O. Kippenhan. Sept. 1953. Decl. Feb. 27, 1957. 26p. Contract AT(11-1)-74. $4.80(ph OTS); $2.70(mf OTS).

A type of linear accelerator is described, and the design parameters are developed in which the accelerating electrodes are quarter-wave transmission lines with the beam passing through a hole in the open end. The advantages of this type of accelerator are discussed relative to its application as an injector for a large high-current accelerator. The tentative goal is an unpulsed beam of 100 ma ion current, an energy of 0.5 Mev, and a diameter as small as is practical but not greater than 5 in. at the exit side of the second quarter-wave electrode. (auth)

7869 LRL-54
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
MTA PROJECT ACCELERATOR QUARTERLY PROGRESS REPORT [FOR] DECEMBER 1952 THROUGH MARCH 1953. Feb. 1954. Decl. Feb. 26, 1957. 34p. Contract AT(11-1)-74. $6.30(ph OTS); $3.00(mf OTS).

The major problem at the present in the operation of the MTA Mark I is the prevention of damage in the short front-end gaps of the accelerator. Runs No. 6 and 7 utilized, respectively, a new No. 1 drift tube of Inconel to minimize damage during sparking and a longer gap length without drift tubes to prevent sparking. Accelerator component development included an ion source, oscillator, vacuum system, and electronic instruments. A study program was initiated to evaluate the relative economic and technical advantages of higher-frequency accelerators than Mark I. (For preceding period see LRL-53). (J.S.R.)

7870 LRL-60
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
DESIGN AND CONSTRUCTION OF THE LIVERMORE LINEAR ACCELERATOR MTA PROJECT. Feb. 1954. Decl. Feb. 27, 1957. 176p. Contract AT(11-1)-74. $27.30(ph OTS); $8.10(mf OTS).

The principles of linear accelerator operation, design

problems encountered, and the design of the Mark I Linear Accelerator are described. (F.S.)

7871 LRL-89

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

MARK I MTA LINEAR ACCELERATOR RUN NO. 6. D. H. Birdsall, P. V. Livdahl, R. L. Spoerlein, C. M. Steele, S. P. Stone, and R. J. Vetterlein. Feb. 1954. Decl. Feb. 25, 1957. 75p. Contract AT(11-1)-74. $12.30 (ph OTS); $4.50(mf OTS).

Run No. 6 of Mark I linear accelerator covers the period of operation from January 21 to March 13, 1953. The run was characterized by a new set of 0 and 1 drift tubes made entirely of Inconel, a modified geometry for drift tube No. 1, and a longer 0–1 gap than previously used. The use of Inconel made it possible to withstand severe damage from sparks and yet return to a point where high rf voltages could be held in the cavity. However, the low thermal conductivity of this material resulted in its cracking so badly from thermal stresses that only short-lived improvement was obtained. Rf voltages as high as 25 Mv end to end of the cavity were obtained without the focusing magnets being energized but with duty cycles as high as 30%. The first beam operation at the design proton voltage of 22.5 Mev was attained. Test targets were bombarded at the design proton rf level, and studies were made of the beam location, energy distribution, and peak-to-average ratio. The bombardment of these targets was made without focusing magnets. The run was finally terminated by sparking in the 0–1 gap and failure of the rf gasket in drift tube No. 7. (For preceding report in series see LRL-81.) (auth)

7872 LRL-114

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

OPERATION OF A HIGH CURRENT RF INJECTOR FOR AN MTA-TYPE ACCELERATOR. W. A. S. Lamb. May 1954. Decl. Feb. 28, 1957. 16p. Contract AT(11-1)-74. $3.30(ph OTS); $2.40(mf OTS).

A high current r-f injector (Ion III), having r-f as well as d-c accelerating stages, has been successfully operated. Proton beams of 450-kev energy and peak currents of 150 milliamperes were obtained in a column approximately three inches in diameter. A description is given of the operation under various conditions, and measurements of the beam energy are described. Alternating gradient strong focusing lens systems were used to control the beam. Measurements were made of the frequency shift of the resonant cavity accelerator due to beam loading which confirm theoretical calculations. (auth)

7873 LRL-125

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

MARK I FOCUSING MAGNETS. R. L. Spoerlein. Apr. 1954. Decl. Feb. 26, 1957. 10p. Contract AT(11-1)-74. $1.80(ph OTS); $1.80(mf OTS).

The MTA Mark I linear accelerator was designed for relatively high particle currents, thus the usual type of focusing employing drift tube grids could not be used, and as an alternate scheme, a solenoidal electromagnet in each drift tube was used. This type of focusing was successful as is evidenced by the fact that reasonably high currents of accelerated particles were measured. However, certain fundamental design criteria limited the variations of the magnetic lens' parameters to the extent that only relative data were obtained. These data and their analyses are presented in this report. (auth)

7874 LRL-131

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

ELECTRIC FIELD MEASUREMENTS IN THE MARK I ACCELERATOR BY THE "GLO-BALL" METHOD. D. H. Birdsall, C. C. Old, and S. P. Stone. May 1954. Decl. Feb. 26, 1957. 13p. Contract AT(11-1)-74. $3.30(ph OTS); $2.40(mf OTS).

A general description of the techniques, a typical axial field plot for Mark I, radial field plots of the critical 0 to 1 gap, and an axial field plot with No. 1 drift tube removed are presented. The over-all accuracy of the method is ±5 per cent which compares favorably with the laborious field perturbation methods. Present data indicate that the "glo-ball" technique is independent of r-f magnetic fields and that it is practical at all r-f frequencies provided the cavity is of adequate size to accommodate the ball. (auth)

7875 LRL-132

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

ELECTRIC FIELD INVESTIGATIONS WITH THE 1/10 SCALE RF MODEL OF MARK I. J. F. Steinhaus. May 1954. Decl. Feb. 26, 1957. 29p. Contract AT(11-1)-74. $4.80(ph OTS); $2.70(mf OTS).

Investigations made of different drift tube contours, and their effects on the electric field distribution along the longitudinal axis in the $1/10$ scale r-f model of the Mark I MTA linear accelerator are reported. Electric field plots were obtained for different drift tube types and different drift tube positions. The data were analyzed to determine the best electrical operating conditions for an accelerator from the point of view of voltage holding ability of the loaded cavity and its beam dynamics. Results of model testing led to the adoption of the original design for drift tube No. 1, and a new drift tube gap arrangement for the Mark I accelerator. Only relatively large changes in drift tube positions caused excessive stem currents. Results pertaining to electric field measurements were within ±5% of those obtained by the "glo-ball" method on the full scale Mark I machine. (auth)

7876 LRL-147

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

MTA PROJECT ACCELERATOR QUARTERLY PROGRESS REPORT FOR OCTOBER THROUGH DECEMBER 1953. June 1954. Decl. Feb. 28, 1957. 33p. Contract AT(11-1)-74. $6.30(ph OTS); $3.00(mf OTS).

The performance of the Mark I MTA was improved during run no. 10, by installing grid-driven PA tubes in the r-f system, replacing the self-excited oscillators. Faster cavity voltage recovery (a few milliseconds) was obtained, which made the r-f on-time approach 100%. Faster pump-down speeds and higher peak beam currents were also achieved. Relative neutron flux and beam current production data, as a function of injection energy, are presented. Higher current beam levels were limited only by the capacity of the beam injector. Design of the new A-54 50-mc accelerator is beginning. Ion source development for this machine is aimed at eliminating the short drift-tube gaps incorporated in Mark I. Theoretical analyses indicate that solenoidal rather than strong-focusing magnet design will be more feasible for this accelerator. (K.S.)

7877 LRL-159

California Research and Development Co. Livermore Research Lab,, Livermore, Calif.

MTA PROJECT ACCELERATOR TERMINAL REPORT [FOR] JANUARY THROUGH JUNE 1954. June 1954.

Decl. Feb. 26, 1957. 10p. Contract AT(11-1)-74. $1.80 (ph OTS); $1.80(mf OTS).

The design and construction status of the high-frequency A54 linear accelerator is reported as of the termination of the CR & D technical effort. (K.S.)

7878 LWS-24318

California Research and Development Co., Livermore, Calif.

BEAM PROFILE. N. C. Ostrander. Feb. 15, 1952. Decl. Feb. 28, 1957. 12p. (CRD-T-1-78). $3.30(ph OTS); $2.40(mf OTS).

The necessary and sufficient condition for beam profile invariance with distance is a uniform distribution in phase. If this condition is not satisfied, then the intensity at the beam center will fluctuate with distance. A distribution in amplitude alternative to that used by Panofsky is investigated. The revised distribution is based upon energy considerations and indicates a relatively higher intensity beam center than earlier calculations. A means of artificially introducing a non-uniformity in phase distribution is suggested. This will reduce the high intensity beam center to such an extent that the modified profile appears to be much more favorable than that of the rotational beam currently being used in engineering calculations. (auth)

7879 MTA-4(Rev.)

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

TEMPERATURE AND THERMAL STRESS DISTRIBUTION IN CLAD PLATES. N. R. LeRoy, R. L. McKisson, J. P. Frankel, and G. W. Brown. Dec. 8, 1952. Revised June 1953. Decl. Mar. 7, 1957. 21p. Contract AT(11-1)-74. $0.30(OTS).

Equations have been derived for computing the steady state temperature and thermal stress distribution in thin clad plates free of external restraint and in which there is uniform volumetric heat generation. Since in many cases the clad plates are in an unknown prestressed condition before operation, a method has been outlined for estimating this residual stress and combining it with the thermal stresses to obtain the operating stress distribution. The results when applied to the Zr-clad U plate indicate a steady state operating stress of 11,400 psi compression in the cladding and a maximum of 8,600 psi tension in the U core. These values are considerably lower than the residual stresses at room temperatures. Therefore it is concluded that if clad plates can be successfully fabricated, the steady state thermal stresses themselves should not cause failure. (auth)

7880 NAA-SR-156

North American Aviation, Inc., Downey, Calif.

BEAM SNOUTING AND ITS APPLICATION TO THE MTA MARK III. S. Cohen. Oct. 31, 1951. 26p. Decl. Mar. 11, 1957. 26p. Contract AT-11-1-gen-8. $4.80(ph OTS); $2.70(mf OTS).

The possibility of using certain electromagnetic methods for confining a beam of charged particles to a long tube is considered. It is shown that methods involving no axial components of the magnetic fields result in no net confinement to first order but do cause a shaping of the beam, while methods involving axial components do produce such a confinement. In particular, magnetic lenses and long solenoids are considered and power requirements for application to the MTA Mark III are estimated. Suggestions are made for the most feasible method for use with the MTA Mark III. (auth)

7881 UCRL-1686

California. Univ., Berkeley. Radiation Lab.

MODES OF ACCELERATION OF IONS IN A 3-DEE CYCLOTRON. M. Jakobson, M. Heusinkveld, and L. Ruby. Feb. 29, 1952. Decl. Mar. 5, 1957. 17p. Contract W-7405-eng-48. $3.30(ph OTS); $2.40(mf OTS).

Analysis shows that a symmetrical three-dee cyclotron accelerating ions below relativistic velocities has the property that ions of differing e/m ratios can be accelerated without changing the frequency of the electrical power supplied to the dees or the value of the magnetic field. The energy gain per revolution of the ions which can be accelerated in the various modes are found, considering three identical dees of arbitrary angular widths. For these calculations it has been assumed that a step-function voltage change occurs at each edge of the dees. (auth)

RADIATION EFFECTS

882 ANL-5395

Argonne National Lab., Lemont, Ill.

RADIATION DAMAGE IN DIAMOND AND SILICON CARBIDE. [PART] II. W. Primak. Mar. 16, 1955. Decl. Feb. 13, 1957. 13p. Contract W-31-109-eng-38. $0.25 (OTS).

Further observations of changes in lattice constants and densities of diamond and silicon carbide on exposure in various nuclear reactors are given and discussed. (auth)

7883 ANL-5477

Argonne National Lab., Lemont, Ill.

EFFECTS OF IRRADIATION ON CAST SPECIMENS OF URANIUM–CHROMIUM EUTECTIC ALLOY. Final Report—Metallurgy Program 6.1.17. J. H. Kittel and S. H. Paine. Dec. 1955. Decl. Feb. 12, 1957. 15p. Contract W-31-109-eng-38. $0.30(OTS).

In order to determine the irradiation behavior of uranium–chromium eutectic alloy (U–5 w/o Cr), which is being considered as fuel material in the proposed Dow-Detroit Edison reactor, cast enriched specimens of the alloy were irradiated in the MTR in NaK-filled capsules to total atom burnups ranging up to 0.65% (5500 Mwd/T). The alloy was irradiated both bare and clad with zirconium, stainless steel, or Globe iron. Although none of the specimens showed large length changes resulting from irradiation, they all developed extreme surface roughening, which could be only partially restrained by the claddings investigated. The general deterioration of the specimens under irradiation apparently resulted from an undesirably large grain size, and it was recommended that future irradiations on the alloy be made only on specimens known in advance to be fine-grained. (auth)

7884 ANL-5537

Argonne National Lab., Lemont, Ill.

PRELIMINARY STUDIES OF IRRADIATION DAMAGE TO URANIUM–ZIRCONIUM ALLOYS. S. H. Paine, F. L. Brown, and W. F. Murphy. Oct. 1956. Decl. Feb. 12, 1957. 28p. Contract W-31-109-eng-38. $0.35(OTS).

The addition of Zr to U is shown to have a beneficial effect upon the irradiation stability of the fuel alloy. Preliminary methods used to discover this fact, and experimental problems encountered are described. (auth)

7885 CF-49-4-141

Oak Ridge National Lab., Tenn.

GAMMA IRRADIATION OF OILS AND PLASTICS. R. N. Lyon. Apr. 13, 1949. Decl. Feb. 7, 1957. 5p. $1.80(ph OTS); $1.80(mf OTS).

Five oils and three plastics, fluorothene, polythene, and saran, were subjected to a total of 1.4×10^7r of γ radiation over a period of one month. The results of these tests are tabulated. (B.J.H.)

7886 HW-29987

[Hanford Atomic Products Operation, Richland, Wash.]

EXPOSURE OF TWO DIFFERENT TYPES OF RUBBER TO PILE WATER. Final Report—Production Test No. 105-505-S1. J. P. Cooke. Nov. 18, 1953. Decl. Feb. 20, 1957. 4p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80 (mf OTS)

Tests were made on the deterioration rates of two rubbers in reactor water in order to determine the feasibility of their use as seals in reactor equipment. Both the neoprene and natural rubber suffered extremely severe deterioration during a 30 day exposure. (B.J.H.)

7887 MonP-208

Clinton Labs., Oak Ridge, Tenn.

EFFECTS OF RADIATIONS ON MATERIALS. F. C. Armistead. Nov. 11, 1946. Decl. Mar. 8, 1957. 25p. Contract W-7405-eng-39. $4.80(ph OTS); $2.70(mf OTS).

The effects of reactor radiations on properties of reactor materials, including fuels, are reviewed. (C.W.H.)

7888 NAA-SR-909

North American Aviation, Inc., Downey, Calif.

RADIATION EFFECTS QUARTERLY PROGRESS REPORT [FOR] OCTOBER–DECEMBER 1953. F. W. Faris, ed. Feb. 22, 1954. Decl. Mar. 4, 1957. 46p. Contract AT-11-1-GEN-8. $0.40(OTS).

The effects of neutron bombardment on various properties of graphite were measured, and an evaluation was made of the a-c method for measuring the magneto-resistivity coefficient of graphite. A comparison was made of radiation damage to graphite in Brookhaven and in Hanford reactors; the role of ionizing radiation in radiation annealing was investigated, and a theoretical study was made of the effect of neutron bombardment on the low-temperature thermal conductivity of graphite. Data are reported from a comparison of the relative disordering rates of 36-Mev α particles and 9-Mev protons on a specimen of Cu_3Au; a study of the effects of temperature gradients of diffusion in crystals; thermoelectric studies on Cu; and the effects of radiation on shear modulus and internal friction in Cu. An estimate was made of the heat of reaction for the process in which colloidal particles of K embedded in a KCl crystal are thermally dissociated into normal K ions and an equal of F-centers in the crystal lattice. Studies were made of optical effects in alkali halides. Modifications are described in an apparatus for measuring the thermoelectric power of graphite in place on the cyclotron. Cyclotron operation is discussed, and minor design changes are described. (C.H.)

7889 NAA-SR-1286

North American Aviation, Inc., Downey, Calif.

RADIATION EFFECTS QUARTERLY PROGRESS REPORT [FOR] OCTOBER–DECEMBER 1954. Frank E. Faris and Theodore R. Johnson, eds. July 15, 1955. Decl. Mar. 1, 1957. 48p. Contract AT-11-1-GEN-8. $0.40(OTS).

A summary is presented of the results of studies of the temperature variation of the magnetic susceptibility of graphite, a quantitative interpretation of neutron damage in graphite, and the effects of bromination of graphite. The susceptibility of graphite has been determined as a function of temperature from 20 to 300°K. The changes in electrical resistivity, thermoelectric power, and thermal conductivity of proton and neutron irradiated graphite have been determined as a function of annealing temperature. The stored energy release from irradiated graphite is shown graphically as a function of temperature rise. A new computational procedure for determining the density of states in the cubic structures of Th, U, and Pu is outlined. The thermal conductivity of a single crystal of U was measured and found to be not significantly different from that of a polycrystalline sample. An anomalous increase in electrical resistivity has been observed in proton-deutron irradiated U when annealed. The magnetic susceptibility of Th is being studied. A method of zone melting U and Th is being developed utilizing the "Kaufman" arc furnace principle. Values for the vacancy formation energy and vacancy concentration near the melting point of Al have been determined from thermal expansion coefficient data. The field dependence of the magnetoresistance and Hall potential of Bi have been studied. The effect of annealing subsequent to electron bombardment on the shear modulus of Cu is being studied. A new cyclotron beam profile indicator consisting of 6 Ta probes has been in operation with satisfactory results. The effects of impurity He atoms in U are being studied. (M.P.G.)

SHIELDING

7890 HW-24258

[Hanford Works, Richland, Wash.]

TEMPERATURE DISTRIBUTION IN THE CONCRETE SHIELD. John R. Triplett. Apr. 28, 1952. Decl. Feb. 27, 1957. 12p. Contract [W-31-109-Eng-52]. $3.30(ph OTS); $2.40(mf OTS).

The temperature in the Brookhaven concrete shield proposed for the X reactor has been calculated on the basis of no cooling tubes, in order to estimate the effective cooling due to the gun barrels. This is found to be a minor effect, most of the heat flow going to the outer face of the shield. (auth)

7891 KAPL-M-LHW-5

Knolls Atomic Power Lab., Schenectady, N. Y.

EFFECT OF BORON IN REDUCING THERMAL NEUTRON CAPTURES IN IRON. L. H. Weinberg. Feb. 16, 1955. Decl. Feb. 23, 1957. 10p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

The combination of steel followed by a hydrogenous material is frequently used in neutron shielding. This may produce a serious source of hard (~7 Mev) gamma rays due to neutrons which become thermalized in the hydrogenous material and return to the steel to be captured. This experiment gives an indication of the reduction in the thermal neutron flux near the boundary due to the presence of boron in the moderator, boron at the steel moderator boundary, and boron in the moderator and at the boundary. (auth)

URANIUM AND URANIUM COMPOUNDS

7892 ANL-4955

Argonne National Lab., Lemont, Ill.

EFFECTS OF PREFERRED ORIENTATION AND GRAIN SIZE ON DIMENSIONAL STABILITY OF URANIUM ON THERMAL CYCLING AND IRRADIATION. Final Report-Metallurgy Program 5.1.7. H. H. Chiswik. Mar. 1956. Decl. Feb. 11, 1957. 57p. Contract W-31-109-Eng-38. $0.50(OTS).

The growth of alpha-rolled U rods on thermal cycling has been shown to depend on both preferred orientation and grain size. Preferred orientation appears to be a necessary condition for growth to occur; the extent of the growth depends upon the sharpness and type of texture developed and on the grain size. The highest growth rates occur in specimens with highly developed textures coupled with small grain sizes. The growth rates increase with cycling level, particularly in specimens of large grain size. On irradiation the grain size appears to be of no special significance insofar as longitudinal growth is concerned; the primary variable controlling growth on irradiation is

the preferred orientation. As in thermal cycling, increasing the degree of preferred orientation results in increased growth. The lack of correlation between thermal cycling and irradiation as regards the grain size variable suggests different mechanisms for the growth in the two processes: one involving grain boundary interactions, and the other dependent primarily on some basic property of the grains themselves. Room temperature thermal expansion coefficients were found to be good qualitative guides in evaluating the comparative growth behavior: material highly anisotropic in expansion and possessing low expansion coefficients in the rolling direction grew more than specimens of lower anisotropy, in both irradiation and thermal cycling. When applied to thermal cycling, however, consideration must also be given to grain size effects: in thermal cycling a low expansion coefficient in the rolling direction does not necessarily signify a high growth rate since the grain size may be large, which would tend to reduce the growth rate. (auth)

7893 ANL-5029

Argonne National Lab., Lemont, Ill.

EFFECTS OF METAL PURITY AND HEAT-TREATMENT ON THE CORROSION OF URANIUM IN BOILING WATER. J. E. Draley and J. W. McWhirter. Apr. 14, 1953. Decl. Feb. 12, 1957. 19p. Contract W-31-109-eng-38. $0.30(OTS).

Corrosion rates of present reactor-grade U were measured in boiling distilled H_2O and were found to have higher values almost by a factor of two than previously reported corrosion rates of U. Mallinckrodt biscuit metal showed corrosion rates in the same medium somewhat lower than reactor-grade U, and high-purity metal prepared at Argonne National Lab. corroded considerably less rapidly than the biscuit metal. Both reactor-grade metal and biscuit metal showed improvement in corrosion resistance as a result of heating into the β or, more effectively, the γ region and H_2O quenching. In one test, furnace cooling from 1000°C gave the reactor-grade metal its lowest corrosion rate. The high-purity metal showed no such sensitivity to heat treatment but corroded at essentially the same rate regardless of thermal history. The possibility is raised that the reduction of the corrosion rate of less pure U by heat treatment is the result of the solubilizing and retention or redistribution of impurities with respect to the matrix phase. The only impurity indicated to be important by the evidence at hand is Fe. (auth)

7894 ANL-5124

Argonne National Lab., Lemont, Ill.

COMPARATIVE ANALYSIS OF ANL HIGH PURITY URANIUM. B. Blumenthal and H. H. Chiswik. Sept. 24, 1953. Decl. Feb. 11, 1957. 6p. Contract W-31-109-Eng-38. $1.80(ph OTS); $1.80(mf OTS).

The chemical and spectrographic results from a comparative analysis of ANL high-purity U conducted by various laboratories are given. (J.A.G.)

7895 ANL-5214

Argonne National Lab., Lemont, Ill.

DIMENSIONAL STABILITY OF URANIUM POWDER COMPACTS UPON THERMAL CYCLING. R. M. Mayfield, S. T. Segler, and H. H. Chiswik. Nov. 30, 1953. Decl. Feb. 12, 1957. 32p. Contract W-31-109-eng-38. $0.35 (OTS).

Results indicate that compacts of nearly theoretical density and fine grain size can be obtained by hot pressing U or uranium hydride powders in the high α temperature range. Large diameter compacts (1.4 in. in diameter) of varying length are best made by hot pressing in a single operation or by sinterwelding of individual hot pressed components. The magnitudes of growth or shrinkage of such compacts upon thermal cycling approximate those of beta-treated rolled U; the surface wrinkling problem, however, associated with beta treatment is practically eliminated in the alpha pressed powder compacts. (auth)

7896 ANL-5350

Argonne National Lab., Lemont, Ill.

EFFECTS OF ALLOYING ELEMENT ADDITIONS OF MOLYBDENUM, SILICON, ALUMINUM, TITANIUM, VANADIUM, AND NIOBIUM ON THE THERMAL CYCLING STABILITY OF URANIUM. Final Report of Programs 5.1.3 and 5.1.5. Work completed: December 1950. S. T. Zegler and H. H. Chiswik. May 29, 1956. Decl. Feb. 14, 1957. 34p. Contract W-31-109-eng-38. $0.35(OTS).

Results seem to indicate that alloying additions in suitable amounts offer a means of alleviating the surface roughening problem, but that their role in restricting the longitudinal growth is relatively minor. The 600°C rolled alloy rods grew appreciably in a manner similar to unalloyed U, although at somewhat different rates; to restrict their growth, quenching from beta or gamma phase temperatures was necessary, in much the same way as for unalloyed material. In the beta- and gamma-treated conditions some of the alloys shrank slightly as compared with the slight elongations noted in unalloyed uranium similarly treated. The results derived from this study are summarized as follows: alloying element additions of 0.79 wt. % Mo, 0.07 wt. % Si, 0.56 wt. % Al, 0.35 wt. % Ti, and 0.33 wt. % V were ineffective in eliminating the growth of 600°C rolled rods; the above alloying element additions decreased the surface roughness after quenching from beta-phase or gamma-phase temperatures, but to eliminate surface roughening completely a concentration of about 2 wt. % alloying element was needed in the case of Mo; and the as cast U–2 wt. % Mo specimens and U–0.6 wt. % Nb specimens showed essentially no longitudinal growth and no surface roughening on thermal cycling. (auth)

7897 BMI-259

Battelle Memorial Inst., Columbus, Ohio.

USE OF ELECTRICAL MEASUREMENTS TO DETECT DIFFERENCES AMONG URANIUM TETRAFLUORIDE SAMPLES. Final Report. D. C. Reynolds and A. E. Middleton. Apr. 15, 1952. Decl. Feb. 13, 1957. 11p. Contract AT(30-1)-Gen-228. $3.30(ph OTS); $2.40(mf OTS).

Experimental work on the utilization of electrical measurements to detect chemical and physical differences in samples of uranium tetrafluoride is reported. An attempt was made to correlate differences in electrical properties of UF_4 with variations in yield of uranium metal in production bombs. (auth)

7898 BMI-1009

Battelle Memorial Inst., Columbus, Ohio.

CERAMIC INVESTIGATIONS OF UO_2. Adrian G. Allison and Winston H. Duckworth. June 16, 1955. Decl. Feb. 13, 1957. 32p. Contract W-7405-eng-92. $0.35(OTS).

A UO_2 powder with unusually good sinterability was prepared from MCW UO_3 by milling it for 4 hr in 0.02N H_2SO_4 or in distilled water and then reducing it to UO_2 in water-saturated H_2 at temperatures of 930 to 1110°F. Compacts of these powders, pressed at 20,000 psi and sintered at 2800°F for 1 hr in H_2, had bulk densities of 10.75 to 10.93 g per cm^3. As-received MCW UO_2 powder was made free flowing by mechanical agglomeration, using camphor as a temporary binder. The agglomerated powders had good pressing characteristics, and the treatment did not appear to affect the sintering behavior or the resistance of sintered compacts to corrosion by 650°F water or 750°F steam. Twelve simulated fuel elements were made for in-pile tests. Plus 325-mesh MCW UO_2 was hydrostatically

pressed into core rods at 100,000 psi and sintered in H_2 at 3750°F. Bulk densities of the cores varied from 9.71 to 9.96 g per cm.^{3} The sintered UO_2 cores were ground to the desired diameter and length, and were clad in Zircaloy-2 tubes by an arc-welding process. Preliminary investigation indicated ceramic extrusion to be a satisfactory method for fabricating UO_2 fuel-element cores, providing that the UO_2 powder used is a readily sinterable type. Using such a powder and a quantity of organic plasticizer that introduced less than 0.2 w/o ash, extrusions $^3/_{16}$ in. in diameter had bulk densities of from 9.85 to 10.65 g per cm^3 after sintering at 3000°F in H_2 for 1 hr. The bulk density of compacts made of MCW UO_2 pressed at 100,000 psi and sintered in H_2 at 3200°F was increased from 9.50 to 10.38 g per cm^3 by milling the as-received UO_2 to a maximum particle size of 5 μ in a rubber-lined mill, using U balls. Small lots, up to 3000 g, of UO_2 were fused in an electric-arc furnace using a W-tipped electrode with a water-cooled Cu crucible and a protective atmosphere of pure A. About 5 w/o of the UO_2 was deposited on the cool walls of the furnace by vaporization during the melting operation. About 2000 ppm of W was picked up by the UO_2 during the fusion. Compacts having a sintered bulk density of 10.03 g per cm^3 were made by milling the fused oxide to an average particle size of 7.1 μ, hydrostatically pressing at 100,000 psi, and sintering for 2 hr in H_2 at 3750°F. Between 0.1 and 1 w/o of Zr was picked up during milling of the fused UO_2, using a rubber-lined mill and Zircaloy-2 balls. (auth)

7899 CF-51-10-133
Oak Ridge National Lab., Tenn.
THE OXIDES OF URANIUM. J. M. Fulmer. Oct. 16, 1951. Decl. Feb. 14, 1957. 30p. [Contract W-7405-eng-26]. $0.35(OTS).

7900 CF-53-11-156
Oak Ridge National Lab., Tenn.
THE EXCER PROCESS FOR THE CONVERSION OF URANYL NITRATE TO URANIUM TETRAFLUORIDE. F. R. Bruce. Nov. 25, 1953. Decl. Feb. 21, 1957. 6p. Contract [W-7405-eng-26]. $0.25(OTS).

The Excer Process for the preparation of UF_4 from aqueous $UO_2(NO_3)_2$ is reviewed. Uranyl nitrate is converted to UO_2F_2 by ion exchange; then, UO_2F_2 is electrolytically reduced to hydrated UF_4. Advantages of the Excer Process include adequate fission-product decontamination, production of UF_4 at a lower cost, and better separation of U from ionic impurities; the main disadvantage results from the handling of aqueous HF and its associated corrosion problems. Cost estimates are included. (C.W.H.)

7901 CF-54-9-89
Oak Ridge National Lab., Tenn.
NUCLEAR SAFETY TESTS ON A PROPOSED BALL MILL. L. W. Gilley and Dixon Callihan. Sept. 14, 1954. Decl. Feb. 14, 1957. 6p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

A series of critical experiments were run, using UO_2F_2 solution in a 15 in.-diameter stainless-steel reactor which was effectively reflected with H_2O, in an attempt to evaluate the nuclear hazards for a ball mill for enriched U. Steel rods were substituted for the balls as a poison. (F.S.)

7902 HW-28309
[Hanford Atomic Products Operation, Richland, Wash.]
ELECTRICAL RESISTANCE MEASUREMENTS OF AN IRRADIATED URANIUM SAMPLE. Robert S. Kemper and C. L. Boyd. June 8, 1953. Decl. Feb. 19, 1957. 9p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

Electrical resistance data were obtained through the temperature range 15 to 80°C for three U specimens. Two of these samples were alpha-rolled, lead-dip canned uranium, one of which was prepared from a slug that had received 150 Mwd/T exposure in a Hanford process tube. The third sample was unirradiated normal beta heat-treated production material. An approximate 4% increase in electrical resistivity at 20°C was noted after irradiation. The temperature coefficient of electrical resistance was found to be slightly lower for the irradiated material. The resistivity of the alpha-rolled, lead-dip canned, non-irradiated material is approximately 5% higher than the β heat-treated U which is indicative of orientation effects. (auth)

7903 KAPL-1102
Knolls Atomic Power Lab., Schenectady, N. Y.
OBSERVATIONS ON THE EFFECT OF CARBON IN URANIUM SUBJECT TO THERMAL CYCLING AND PILE IRRADIATION. W. E. Seymour and J. F. Duffey. July 14, 1954. Decl. Feb. 26, 1957. 31p. Contract W-31-109-Eng-52. $0.30(OTS).

Carbon in amounts of 0.2 wt. % and over has been observed to improve the dimensional stability of U subject to thermal cycling. It has also been observed to increase the corrosion resistance of U in boiling water and to result in a finer alpha grain size after recrystallization. Carbon does not seem to improve the dimensional stability of U exposed to a neutron flux, although high alpha recrystallization of cold-worked U improves the dimensional stability by a ratio of 5 : 1 upon pile irradiation. (auth)

7904 NAA-SR-64
North American Aviation, Inc., Downey, Calif.
DIFFUSION OF URANIUM CARBIDE IN GRAPHITE. R. L. Loftness. Aug. 2, 1950. Decl. Mar. 9, 1957. 24p. Contract AT-11-1-GEN-8. $0.35(OTS).

The diffusion of UC_2 in graphite at high temperatures was investigated. A typical value found for the diffusion constant D is 8.9×10^{-10} cm^2/sec at 2600°K. The activation energy was determined to be E = 53,000 cal/gm atom, in the temperature range 2400 to 2900°K. Comparison of the experimentally determined values with those of other diffusion systems indicates that the diffusion is primarily a volume phenomenon. Calculation based on the observed rate shows that diffusion of UC_2 fuel has practically no effect on the operation of a high-temperature reactor, even after 1000 hours at 2000°C. (auth)